SUMMARY OF EVALUATION FOR TYPE BU-J PACKAGE ON IAEA SS-6 1985 EDITION (AS AMENDED 1990) (1/16)

Para. No. and Requirement in IAEA SS-6 (1985 edition)	Methodology of Evaluation and Para. in SAR	Result and Conclusion	Remarks
General Requirements for All Packagings and Packages			
505. The package shall be so designed in relation to its mass, volume and shape that it can be easily and safely handled and transported. In addition, the package shall be so designed that is can be properly secured in or on the conveyance during transport. (Handling / tie-down)	It is explained that this package has no slinging device and it can be handled while it is placed on a pallet. (b)-A.4.4	It is so designed that it can be easily handled with a fork-lift truck.	
506. The design shall be such that any lifting attachments on the package will not fail when used in the intended manner and that, if failure of the attachments should occur, the ability of the package to meet other requirements of these Regulations would not be impaired. Assessment shall take account of appropriate safety factors to cover snatch lifting. 507. Attachments and any other features on the outer surface of the package which could be used to lift it shall be designed either to support its mass in accordance with the requirements of para. 506 or shall be removable or otherwise rendered incapable of being used during transport. (Strength at lifting)	Not applicable because this package has no slinging device. (b)-A.4.4	-	
508. As far as practicable, the packaging shall be so designed and finished that the external surfaces are free from protruding features and can be easily decontaminated. (Easy to decontaminate)	It is explained that the outer container is a drum made of carbon steel. (a)-C(5)	The surface of the outer container is steel sheet coated with paint and has no unnecessary structural protrusions. Thus its structure is easy to decontaminate.	
509. As far as practicable, the outer layer of the package shall be so designed as to prevent the collection and the retention of water. (Prevention of water capture)	Its prototype was subjected to the water spray test given in paragraph 621. (b)A.5.2	No water intrusion into the outer container was found after one hour long water spray equivalent to about 110mm/hr in the prototype test.	
510. Any features added to the package at the time of transport which are not part of the package shall not reduce its safety.	Explanation is made about the state of the package being tied down. (b)-A.4.4	The package is not influenced by its being tied down.	

SUMMARY OF EVALUATION FOR TYPE BU-J PACKAGE ON IAEA SS-6 1985 EDITION (AS AMENDED 1990) (2/16)

Para. No. and Requirement in IAEA SS-6 (1985 edition)	Methodology of Evaluation and Para. in SAR	Result and Conclusion	Remarks
511. The package shall be capable of withstanding the effects of any acceleration, vibration or vibration resonance which may arise under conditions likely to be encountered in routine transport without any deterioration in the effectiveness of the closing devices on the various receptacles or in the integrity of the package as a whole. In particular, nuts, bolts, and other securing devices shall be so designed as to prevent them from becoming loose or being released unintentionally, even after repeated use. (Acceleration, vibration)	The strength of the bolts to fasten the lid of the inner container against the vertical vibration and horizontal direction during transport was assessed by calculations. (b)-A.4.7	No damage occurs to the fastening bolts of the inner container due to bending, shearing etc. caused by vibratory acceleration during transport.	Calculational assessment is made using the 2G acceleration in the directions, up and down as well as to-and-fro.
512. The materials of the packaging and any components or structures shall be physically and chemically compatible with each other and with the radioactive contents. Account shall be taken of their behavior under irradiation. (Physical or chemical compatibility of material)	All the dissimilar materials in mutual contact within the content, inner container and outer container are listed and it is indicated that they are not interactive physically and/or chemically. (b)-A.4.1	The materials used in this package are chemically stable ones and do not react physically or chemically as a result of their contact with different kind of materials.	
513. All valves through which the radioactive contents could otherwise escape shall be protected against unauthorized operation. (Malfunction of valve)	It is explained that this package has no valve. (a)C6	No instance of valve opening due to its malfunction occurs because the package has no valves.	
514. For radioactive material having other dangerous properties, see para. 407.	Not applicable. Material has no other dangerous properties.	-	The uranium dioxide powder and pellets that are the radioactive material as the content of the package are stable substance.
Requirements on the Type A Packages			
524. Type A packages shall be designed to meet the requirements specified in paras. 505-514 and, in addition, the requirements of paras. 515-517 if carried by air, and of paras. 525-540.	No virtual requirement is set forth in paragraph 524.	-	The package in question will not be carried by aircraft.
525. The smallest overall external dimension of the package shall not be less than 10 cm.	The external dimensions of the package were presented. (a)-C(5)	The package is cylindrical in shape. Each of the sides is not less than 10cm.	Diameter approx. 61 cm Height approx. 88 cm
526. The outside of the package shall incorporate a feature such as a seal, which is not readily breakable and which, while intact, will be evidence that it has not been opened. (Seal)	The seal on the package was explained. (a)-C(5)	The package is given a seal on its junction after the bolts on the lid of the outer container have been tightened. Thus, if the container is opened it will be evidenced by the seal.	

SUMMARY OF EVALUATION FOR TYPE BU-J PACKAGE ON IAEA SS-6 1985 EDITION (AS AMENDED 1990) (3/16)

Para. No. and Requirement in IAEA SS-6 (1985 edition)	Methodology of Evaluation and Para. in SAR	Result and Conclusion	Remarks
527. Any tie-down attachments on the package shall be so designed that, under both normal and accident conditions, the forces in those attachments shall not impair the ability of the package to meet the requirements of the Regulations.	The fact that the package in question does not have any tie-down device was explained and the situation of the package as tied down was explained. (b)-A.4.5	-	The analysis of the tie-down strength when the package is housed in marine transport container is given in the reference data (b)A-2.
528. The design of the package shall take into account temperatures ranging from −40°C to 70°C for the components of the packaging. Special attention shall be given to freezing temperatures for liquid contents and to the potential degradation of packaging materials within the given temperature range. (Temperature range)	The temperature of the package under the solar radiation was derived by calculation and assessment was made using whichever is higher when it was compared with the higher temperature requirement on the component parts of the type A in regard to the maximum temperature. In regard to the minimum temperature -40°C which is the temperature requirement in the lower temperature range for the type A component materials was used as the temperature for assessment although the minimum temperature of the package during its use is -20°C. (b)-B.4.2~3	The higher temperature assessment used 70°C because the maximum temperature under the solar radiation shown in the standard turned out to be 64ßC. The lower temperature assessment used -40°C. The component parts of the package will not have crack, break, etc. in this temperature range.	
529. The design, fabrication and manufacturing techniques shall be in accordance with national or international standards, or other requirements, acceptable to the competent authority.	The applicable standards were explained regarding the main component materials of packaging. (c)-A.1~2	The main materials of the packaging are in accordance with the Japanese Industrial Standards (JIS) and the American National Standards Institute(ANSI) standards.	The outer container complies with JIS Z1600 and the inner container complies with ANSI MH2.14.
530. The design shall include a containment system securely closed by a positive fastening device which cannot be opened unintentionally or by a pressure which may arise within the package.	Comparative assessment is made on the internal pressure of the inner container as a containment system and its pressure resistance regarding the change in temperature and change in pressure expected to occur during transport. (b)-A.4.6	The internal pressure of the inner container is far less than its withstanding pressure resistance even if 70°C is used as the assessment temperature in the higher temperature range of the package. The pressure difference between the inside and outside of the inner container due to a drop of the ambient pressure is also far less than the withstanding pressure. Thus the integrity of the inner container as the hermetic seal boundary is kept intact and the airtightness is maintained.	The assessment about the ambient pressure used 25kPa.

Para. No. and Requirement in IAEA SS-6 (1985 edition)	Methodology of Evaluation and Para. in SAR	Result and Conclusion	Remarks
531. Special form radioactive material may be considered as a component of the containment system.	This assessment is not made because the package is not deemed as a special form.	-	
532. If the containment system forms a separate unit of the package, it shall be capable of being securely closed by a positive fastening device which is independent of any other part of the packaging.	The inner container as a containment system is fixed inside the package and it cannot be separated, so this is not applicable. (a)-C(5)	-	
533. The design of any component of the containment system shall take into account, where applicable, the radiolytic decomposition of liquids and other vulnerable materials and the generation of gas by chemical reaction and radiolysis. (Pressurization of containment system)	Uranium dioxide powder and pellets placed in the containment system are stable and do not put forth gas and the like. Therefore no such assessment is made.	-	
534. The containment system shall retain its radioactive contents under a reduction of ambient pressure to 25 kPa (0.25 kgf/cm2). (Drop of ambient pressure)	Comparative assessment was made on the strength of the inner container serving as the hermetic seal boundary and the differential pressure due to a drop of the ambient pressure in regard to 25kPa ambient pressure. (b)-A.4.6	No leak of radioactive material ensues because it was verified by testing that the inner container can withstand the minimum of 98kPa pressure although the differential pressure of the inner container due to a drop of the ambient pressure reaches 76kPa.	
535. All valves, other than pressure relief valves, shall be provided with an enclosure to retain any leakage from the valve.	The package has no valve. Hence, no enclosure to retain any leakage from valve is necessary. (a)-C6	-	
536. A radiation shield which encloses a component of the package specified as a part of the containment system shall be so designed as to prevent the unintentional release of that component from the shield. Where the radiation shield and such component within it form a separate unit, the radiation shield shall be capable of being securely closed by a positive fastening device which is independent of any other packaging structure. (Release of containment system)	The package is not provided with any special shield. It is therefore unnecessary to prevent release of containment system. (b)-D.3.1	-	

SUMMARY OF EVALUATION FOR TYPE BU-J PACKAGE ON IAEA SS-6 1985 EDITION (AS AMENDED 1990) (5/16)

Para. No. and Requirement in IAEA SS-6 (1985 edition)	Methodology of Evaluation and Para. in SAR	Result and Conclusion	Remarks
537. A package shall be so designed that if it were subjected to the tests specified in paras 619-624, it would prevent: (a) Loss or dispersal of the radioactive contents; and (b) Loss of shielding integrity which would result in more than a 20 % increase in the radiation level at any external surface of the package. (Containment and shielding as the general test conditions)	Assessment with prototype test and calculational analysis are made regarding each of the tests under the general test conditions. The radioactive material leakage was assessed based on the results of the assessment. Shield calculation with QAD code was made to assess the increase of radiation dose equivalent rate on the surface of the package in consideration of the deformation of the package under the general test conditions. (b)-C.3.1, (b)-D.4	It has been verified under the general test conditions that the integrity of the inner container serving as the hermetic seal boundary of this package is maintained. Hence there will be no leakage of the radioactive content. The radiation dose equivalent rate on the surface of the package is 0.036 mSv/hr; hence, it does not exceed the standard limit. The maximum increase rate of the dose equivalent rate on the surface of the package is 3%.	
Relating to para. 537			
619. The tests are: the water spray test, the free drop test, the stacking test, and the penetration test. Specimens of the package shall be subjected to the free drop test, the stacking test and the penetration test, preceded in each case by the water spray test. One specimen may be used for all the tests, provided that the requirements of para. 620 are fulfilled.	Regarding the general test conditions the assessment was made through prototype test and analysis as shown below.	-	
620. The time interval between he conclusion of the water spray test and the succeeding test shall be such that the water has soaked in to the maximum extent, without appreciable drying of the exterior of the specimen. In the absence of any evidence to the contrary, this interval shall be taken to be two hours if the water spray is applied from four directions simultaneously. No time interval shall elapse, however, if the water spray is applied from each of the four directions consecutively.	See the method for assessment given in para. 621.	-	
621. Water spray test. The specimen shall be subjected to a water spray test that simulates exposure to rainfall of approximately 5 cm per hour for at least one hour.	Prototype test of the water spray test given in para. 621 was conducted. (b)-A.5.2	No water intrusion into the outer container was found after one hour long water spray equivalent to about 110 mm/hr in the prototype test.	

Para. No. and Requirement in IAEA SS-6 (1985 edition)	Methodology of Evaluation and Para. in SAR	Result and Conclusion	Remarks
622. Free drop test. The specimen shall drop onto the target so as to suffer maximum damage in respect of the safety features to be tested. (a) The height of drop measured from the lowest point of the specimen to the upper surface of the target shall be not less than the distance specified in Table XIV for the applicable mass. The target shall be as defined in para. 618. (b) For packages containing fissile material the free drop test specified above shall be preceded by a free drop from a height of 0.3 m on each corner or, in the case of a cylindrical package, onto each of the quarters of each rim. (c) For rectangular fibreboard or wood packages not exceeding a mass of 50 kg, a separate specimen shall be subjected to a free drop onto each corner from a height of 0.3 m. (d) For cylindrical fibreboard packages not exceeding a mass of 100 kg, a separate specimen shall be subjected to a free drop onto each of the quarters of each rim from a height of 0.3 m.	The following three attitudes in a free drop from 1.2 m above the ground were assessed through calculational analysis. (1)Horizontal drop (2)Vertical drop (3)Diagonal (corner) drop In addition, calculational assessment was made on 0.3 m drop distance for each of the corners. Moreover, prototype test was conducted for the horizontal drop and diagonal drop of the 1.2 m test and the findings were compared to the calculation results. (b)-A.5.3	The outer container was distorted by 16.1 mm as a result of the horizontal drop. The volume change of the outer container is not more than 0.01 %. The end of the outer container was deformed by 7.1 mm as a result of the vertical drop. The volume change is not more than 0.01%. The Corner of the outer container was deformed by a maximum of 37 mm as a result of the diagonal drop including the deformation due to the 0.3 m diagonal drop. The volume change of the outer container is 0.13% maximum.	
623. Stacking test. Unless the shape of the packaging effectively prevents stacking, the specimen shall be subjected, for a period of 24 h, to a compressive load equal to the greater of the following: (a) The equivalent of 5 times the mass of the actual package; and (b) The equivalent of 13 kPa (0.13 kgf/cm2) multiplied by the vertically projected area of the package. The load shall be applied uniformly to two opposite sides of the specimen, one of which shall be the base on which the package would normally rest.	Prototype test was made for assessment using 1120 kg load which is equivalent to a load of 5 times the weight of the package. (b)-A.5.4	No deformation was found as a result of the stacking test because the height of the outer container did not change because of the test.	

Para. No. and Requirement in IAEA SS-6 (1985 edition)	Methodology of Evaluation and Para. in SAR	Result and Conclusion	Remarks
624. Penetration test. The specimen shall be placed on a rigid, flat, horizontal surface which will not move significantly while the test is being carried out. (a) A bar of 3.2 cm in diameter with a hemispherical end and a mass of 6 kg shall be dropped and directed to fall, with its longitudinal axis vertical, onto the centre of the weakest part of the specimen, so that, if it penetrates sufficiently far, it will hit the containment system. The bar shall not be significantly deformed by the test performance. (b) The height of drop of the bar measured from its lower end to the intended point of impact on the upper surface of the specimen shall be 1 m.	Prototype test was conducted to assess the situation resulted from the impact of a drop from 1 m above of a steel rod 3.2 cm in diameter weighing 6 kg onto the package. (b)-A.5.5	The exterior sheet of the outer container was not penetrated and the inner container was not affected by the drop of the steel rod as a result of the prototype test regarding the two postures of the package, standing on its bottom and lying on its side.	
538. The design of a package intended for liquid radioactive material shall make provision for ullage to accommodate variations in the temperature of the contents, dynamic effects and filling dynamics.	The requirements for liquid radioactive material are not applicable because the radioactive content of this package is solid uranium dioxide powder and pellets.	-	
539. A Type A package designed to contain liquids shall, in addition: (a) Be adequate to meet the conditions specified in para. 537 above if the package is subjected to the tests specified in para 625; and (b) For packages in which the liquid volume does not exceed 50 ml, be provided with sufficient absorbent material to absorb twice the volume of the liquid contents. Such absorbent material must be suitably positioned so as to contact the liquid in the event of leakage; and (c) For packages in which the liquid volume is greater than 50 ml, either: (i) be provided with sufficient absorbent material as prescribed in subpara. 539(b); or (ii) be provided with a containment system composed of primary inner and secondary outer containment components designed to ensure retention of the liquid contents within the secondary outer containment components, even if the primary inner components leak.	The requirements for liquid radioactive material are not applicable because the radioactive content of this package is solid uranium dioxide powder and pellets.	-	

SUMMARY OF EVALUATION FOR TYPE BU-J PACKAGE ON IAEA SS-6 1985 EDITION (AS AMENDED 1990) (8/16)

Para. No. and Requirement in IAEA SS-6 (1985 edition)	Methodology of Evaluation and Para. in SAR	Result and Conclusion	Remarks
539. (Continued) However, the requirements given in subparas 539(b) and (c) shall not apply in the case of a Type B package designed and approved for liquids which contains the same liquids having an activity equal to or less than the A2 limit for the authorized contents.			
540. A package designed for compressed gases or uncompressed gases shall prevent loss or dispersal of the radioactive contents if the package were subjected to the tests specified in para. 625. A package designed for contents not exceeding 40 TBq (1000 Ci) of tritium or for noble gases in gaseous form with contents not exceeding A2 shall be excepted from this requirement.	The requirements for liquid radioactive material are not applicable because the radioactive content of this package is solid uranium dioxide powder and pellets.	-	
Requirements for packages containing fissile material			
559. Except as provided in para. 560, packages containing fissile material shall be so designed, and used, to comply with the requirements specified in paras 561 - 568 , as well as those specified in paras 518 - 520, 524 or 541, as applicable, taking into account the nature, activity and form of the contents.	Paragraph 559 does not include any virtual requirement. Paragraphs 518 through 520 are intended for the type IP and they are not applicable to the package in question. Paragraph 524 is the requirement for the type A package and it has been explained by the foregoing. Para. 541 is the requirement for the type B package and is not applicable to the package in question.	-	
560. Packages meeting one of the requirements of subparas 560(a) - 560(f) shall be excepted from the requirements specified in paras 561 - 568, and from the other requirements of these Regulations that apply specifically to fissile material; such packages, however, shall be regulated as non-fissile radioactive material packages as applicable, and shall still be subject to those requirements of these Regulations which pertain to their radioactive nature and properties. (a) Packages containing individually not more than 15 g of fissile material, ...	The package in question does not fall under the category defined in para. 560 as fissile material package.	-	

Para. No. and Requirement in IAEA SS-6 (1985 edition)	Methodology of Evaluation and Para. in SAR	Result and Conclusion	Remarks
560.(Continued) (b) Packages containing homogeneous hydrogenous solutions or mixtures....... (c) Packages containing uranium enriched in uranium-235 to a maximum of .... (d) Packages containing not more than 5 g of fissile material in any 10 litre .... (e) Packages containing individually not more than 1 kg of total plutonium, ... (f) Packages containing liquid solutions of uranyl nitrate enriched in			
561. packages containing fissile material shall be transported and stored in accordance with the relevant controls in Section IV.	The package in question is transported in accordance with chapter IV.	-	
562. Fissile material shall be packaged and shipped in such a manner that subcriticality is maintained under conditions likely to be encountered during normal conditions of transport and in accidents. The following contingencies shall be considered: (a) Water leaking into or out of packages; (b) The loss of efficiency of built-in neutron absorbers or moderators; (c) Possible rearrangement of the radioactive contents either within the package or as a result o loss from the package; (d) Reduction of spaces between packages or radioactive contents; (e) Packages becoming immersed in water or buried in snow; and (f) Possible effects of temperature changes.	The requirements in (a) through (f) of para. 562 are incorporated into the criticality assessment of para. 566 (isolated system) and para. 567 (array system).	-	

Para. No. and Requirement in IAEA SS-6 (1985 edition)	Methodology of Evaluation and Para. in SAR	Result and Conclusion	Remarks
563. A packaging for fissile material shall be so designed that, if it were subjected to the tests specified in paras 619 - 624: (a) Neither the volume nor any spacing on the basis of which nuclear criticality control for the purpose of para. 567(a) has been assessed would suffer more than 5 % reduction, and the construction of the packaging would prevent the entry of a 10 cm cube; and (b) Water would not leak into or out of any part of the package unless water in-leakage or out-leakage, to the optimum foreseeable extent, has been assumed for the purposes of paras 566 and 567; and (c) The configuration of the radioactive contents and the geometry of the containment system would not be altered so as to increase the neutron multiplication significantly.	Regarding the general test conditions in paras. 619 through 624, the prototype test and calculational analysis plus others combined were used in assessment. (b)-A.5.7, A.9.1	(a) While the outer container was taken into account by the criticality analysis of non-damaged package and damaged package, the volume of the maximum deformation under the general test conditions was 0.13 % and no dent has been made. Thus no reduction in volume or space in excess of 5 % has occurred and no dent that can allow anything of 10 cm cube in the structure of the packaging has been formed. (b) The assessment has found that water has not entered into the outer container as a result of the water spray test under the general test conditions. No water has been located in the inner container, either. Therefore, no water penetration into the package has occurred, and no water leak from the package has occurred. (c) Under the general test conditions the integrity of the inner container has been kept intact. Thus, the array of the radioactive content and the configuration of the containment system have not changed.	

SUMMARY OF EVALUATION FOR TYPE BU-J PACKAGE ON IAEA SS-6 1985 EDITION (AS AMENDED 1990) (11/16)

Para. No. and Requirement in IAEA SS-6 (1985 edition)	Methodology of Evaluation and Para. in SAR	Result and Conclusion	Remarks
564. For the purposes of the evaluation in this subsection: (a) Undamaged shall mean the condition of the package as it is designed to be presented for transport; (b) Damaged shall mean the evaluated or demonstrated condition of the package if it had been subjected to whichever of the following combination of tests is the more limiting: (i) The tests specified in paras 619 -624 followed by the tests specified in paras 626 - 628 and completed by the tests specified in paras 631 - 633. The mechanical test of para. 627 shall be that required by para. 548. (ii) The tests specified in paras 619 - 624 followed by the test in para. 629.	The following methods were used to assess or demonstrate the state of the package as damaged. (1) General test conditions ①Water spray test Assessment was made through prototype test. ②Free drop test Calculational analysis was made and partial prototype test was made for confirmation including the 0.3 m corner drop test as required on the fissile package. ④Stacking test Assessment was made through prototype test. ⑤Penetration test Assessment was made through prototype test. (2) Special test conditions ①Drop test I: 9 m free drop Calculational analysis was made for assessment of horizontal drop, vertical drop and diagonal drop. ②Drop test II: drop onto protrusion Calculational analysis was made for assessment of horizontal drop and vertical drop. ③Fire test Prototype test was made using the specimens subjected to the drop tests in ① and ② above. Calculational compensation was made regarding the conditions of solar radiation. ④Immersion test	The results of assessment for the general test conditions and special test conditions in succession as those on fissile package are shown below. (1) With respect to the drop test the dimensional change and volume change as deformation were assessed assuming 0.3m+1.2m+9m=10.5m drop in consideration of combined effect with the deformation resulting from the drop test under the general test conditions. While the diagonal drop with the bottom plate facing downward gives the greatest dimensional change the vertical drop with the top plate facing downward gives the greatest extent of deformation (7.08%). Hence the dimensional change in the latter case is used as representing the damaged package. (2) The volume change of the outer container resulting from the drop test II is 2.44%, so the total volume change comprising the drop test I and drop test II is 9.52%. (3) The temperature of the gasket in the inner container was recorded to be 105°C in the prototype test of the fire test. Therefore calculation for compensation regarding the ambient temperature and the extent of deformation of the thermal insulator was made and 125°C was used in the assessment as the maximum temperature of the gasket. (4) The result of assessment for the immersion test is that no water has entered into the inner container. Criticality analysis is made using the above	In the prototype test of the BU-J type the fire test is conducted based on the former specification with the specimens subjected to the 9 m drop test and 1 m drop to protrusion test only.

SUMMARY OF EVALUATION FOR TYPE BU-J PACKAGE ON IAEA SS-6 1985 EDITION (AS AMENDED 1990) (12/16)

Para. No. and Requirement in IAEA SS-6 (1985 edition)	Methodology of Evaluation and Para. in SAR	Result and Conclusion	Remarks
564.(Continued)	Assessment was made through prototype test. Para. 564 (b) has defined the state of damage under the series of requirements in (i) as "after undergoing severer testing" (b)-A.5, A.9	results as the state of the damaged package.	

Para. No. and Requirement in IAEA SS-6 (1985 edition)	Methodology of Evaluation and Para. in SAR	Result and Conclusion	Remarks
565. In determining the subcriticality of individual packages in isolation, it shall be assumed that water can leak into or out of all void spaces of the package, including those within the containment system. However, if the design incorporates special features to prevent such leakage of water into or out of certain void spaces, even as a result of human error, absence of leakage may be assumed in respect of those void spaces. Special features shall include the following: (a) Multiple high standard water barriers, each of which would remain leaktight if the package were damaged (see para. 564); a high degree of quality control in the production and maintenance of packagings; and special tests to demonstrate the closure of each package before shipment; or (b) Other features given multilateral approval.	The maximum weight of uranium dioxide allowed to be loaded in each of the package is defined for each of the enrichments to be 90% of the minimum criticality mass in order to secure the criticality safety even if a single isolated package is so filled with water that the optimum moderation is created there. (b)-E.4.2	Subcriticality will remain even if a single isolated package permits water to come into it because the maximum weight of load in each of the packages is set at 90% of the minimum criticality mass.	
566. The individual package damaged or undamaged shall be subcritical under the conditions specified in paras 564 and 565, taking into account the physical and chemical characteristics including any change in those characteristics which would occur when the package is damaged and with the conditions of moderation and reflection as specified below: (a) For the material within the containment system: the material arranged in the containment system (i) In the configuration and moderation that results in maximum neutron multiplication; and (ii) With close reflection of the containment system by water 20 cm thick (or equivalent) or such greater reflection of he containment system as may additionally be provided by the surrounding material of the packaging; and, in addition (b) If any part of the material escapes from he containment system: that material arranged in (i) The configuration and moderation that results in maximum neutron multiplication; and (ii) With close reflection of that material by water 20 cm thick or equivalent).	See the above.	See the above.	

Para. No. and Requirement in IAEA SS-6 (1985 edition)	Methodology of Evaluation and Para. in SAR	Result and Conclusion	Remarks
567. An array of packages shall be subcritical. A number 'N' shall be derived assuming that if packages were stacked together in any arrangement with the stack closely reflected on all sides by water 20 cm thick (or its equivalent) both of the following conditions would be satisfied: (a) Five times 'N' undamaged packages without anything between the packages would be subcritical: and (b) Two times 'N' damaged packages with hydrogenous moderation between packages to the extent which results in the greatest neutron multiplication would be subcritical.	Regarding the package (damaged and non-damaged) of isolated system, criticality analysis with a 3-dimensional configuration model was made using the KENO IV monte carlo criticality calculational code and the 16-group cross-sectional area set built into this code. The limited number of pieces in transport of the packages in question ("N" as a numeral) is 500. (a)The neutron multiplication factor regarding the non-damaged package was calculated using the following calculational model. ①Water is assumed to exist on the outside of the packaging though no water has entered into it. ②Perfect reflection surface is assumed to exist on the outside of the unit cell comprising the package while the infinite number of the packages in excess of 5 times the limited number of pieces in transport are located in an array. ③The assessment was made on the case where the density of water on the outside of the package as a variable parameter is such that the neutron multiplication factor is brought to its maximum. (b) The neutron multiplication factor regarding the damaged package was calculated using the following calculational model.	(a) The neutron multiplication factor with the non-damaged package is at its maximum when the density of water on the outside of the packaging is 0g/cm^3 and the Keff$\pm 3\sigma$ is 0.825$\pm$0.015. Thus the non-damaged packages in the array system remain subcritical. (b) The neutron multiplication factor with damaged package is at its maximum when the density of water on the outside of packaging is 0 g/cm3 and the keff +/-3 sigma is 0.814 +/- 0.018. Thus the damaged packages in the array system also remain subcritical.	Survey calculation was made regarding the relationship between the enrichment of the uranium in the package and its weight loaded in the package to verify that the neutron multiplication factor is at its maximum when the enrichment is 3.0% and the weight of the uranium dioxide loaded in the package is 89.0kg.

Para. No. and Requirement in IAEA SS-6 (1985 edition)	Methodology of Evaluation and Para. in SAR	Result and Conclusion	Remarks
567.(Continued)	①It is assumed that the outside dimensions of the package have been reduced by 4.1% (8.1% as converted into volume) since its volume is decreased by 8.1% as a result of its drop test. ②Water is assumed to exist on the outside of the packaging though no water has entered into it. ③It is assumed that 1008 packages exceeding the double of the limited number of pieces in transport are located in an array of 12 by 12 in 7 tiers and are surrounded by 30 cm thick water walls. ④The assessment was made on the case where the density of water on the outside of the package as variable parameter is such that the neutron multiplication factor is brought to its maximum. (b)-E.4		
568. In evaluating the subcriticality of fissile material in its transport configuration, the following shall apply: (a) The determination of subcriticality for irradiated fissile material may be based on the actual irradiation experience, taking into account significant variations in composition; (b) For irradiated fissile material of unknown irradiation experience the following assumptions shall be made in determining subcrticality: (i) If its neutron multiplication decreases with irradiation, the material shall be regarded as unirradiated; (ii) If its neutron multiplication increases with irradiation, the material shall be regarded as irradiated to the point corresponding to the maximum neutron multiplication; and	Study on the irradiation history is unnecessary because the content of the packages in question is fresh uranium fuel which has not been irradiated. (a)-D(1)	-	

SUMMARY OF EVALUATION FOR TYPE BU-J PACKAGE ON IAEA SS-6 1985 EDITION (AS AMENDED 1990) (16/16)

Para. No. and Requirement in IAEA SS-6 (1985 edition)	Methodology of Evaluation and Para. in SAR	Result and Conclusion	Remarks
568.(Continued) (c) For unspecified fissile material, such as residues or scrap, whose fissile composition, mass, concentration, moderation ratio or density is not known or cannot be identified, the assumption shall be made in determining subcriticality that each parameter that is not known has the value which gives the maximum neutron multiplication under credible conditions of transport.			

— File

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

RECEIVED
FEB 28 1994
C. M. VAUGHAN

FEB 1 1 1994

STSB:NLO
71-9019

General Electric Company
ATTN: Mr. Charles M. Vaughan
PO Box 780
Wilmington, NC 28402

Dear Mr. Vaughan:

This refers to your application dated December 3, 1993, as supplemented December 14, 1993, December 22, 1993, and January 12, 1994, requesting an amendment to Certificate of Compliance No. 9019, for the Model No. BU-7 package.

In connection with our review, we need the information identified in the enclosure to this letter.

Please advise us within 30 days from the date of this letter when this information will be provided. Additional information requested by this letter should be submitted in the form of revised pages. If you have any questions regarding this matter, we would be pleased to meet with you and your staff. Nancy Osgood is the project manager for our review of your application. Ms. Osgood may be contacted at (301) 504-2459.

Sincerely,

Cass R. Chappell

Cass R. Chappell, Section Leader
Cask Certification Section
Storage and Transport Systems Branch
Division of Industrial and
Medical Nuclear Safety, NMSS

Enclosure: As stated

Structural

1. The buckling analysis of the inner container (supplement dated December 14, 1993) shows a margin of only 15% against buckling under 21 psi external pressure. Section VIII of the ASME Code would require a thicker shell to resist 21 psi external pressure, and Section III of the Code would require a safety margin greater than 15% against buckling. The ASME Code applies to quality vessels, whereas the BU-7 inner container is not designed, fabricated, or inspected to the same standards as vessels which meet the ASME Code. Further, the buckling analysis does not account for possible deterioration of the container during service (note that most of the packages in use are at least 10 years old). Justify that the BU-7 containment system has an adequate margin of safety against buckling. Specify the code or standard used for design of the containment vessel of the BU-7 package. Show that this code or standard allows a margin of safety as small as 15% against buckling, and justify that this code or standard is appropriate to use for the containment system in the Model BU-7 package. Note that the integrity of the containment system is relied upon to ensure criticality safety under accident conditions.

2. For the 30-foot drop test, the BU-7 package was dropped on its top closure ring at approximately 45°. The closure ring was deformed on impact, and there was a slight opening of the drum lid. The subsequent puncture test was performed such that the package lid impacted the pin at a location away from the damaged area. The puncture test does not appear to have been performed in the orientation which would cause maximum damage to the package closure. The performance of the containment system (i.e., the ability of the inner container to exclude water) depends on the condition of the gasket after the fire test. The condition of the gasket after the fire test depends on the drum remaining closed. (Note that the insulating foam is charred all the way to the gasket after the fire test, as shown in Figures 35 and 36 of Appendix B of the application.) Justify that the 30-foot drop and puncture tests were performed in the most damaging orientation with respect to maximizing damage to the closure from the puncture test, and subsequently to the gasket from the fire test. Alternatively, perform additional 30-foot drop, puncture, and fire tests of the BU-7 package. The 30-foot drop and puncture tests should be performed in the orientation which produces maximum cumulative damage to the package closure.

3. The application (supplements dated December 14 and 22, 1993) discusses hydrostatic tests that were performed on BU-7 and BU-J packages. The application is not clear with respect to the details of the tests. Revise the application to clearly address the following:

   (a) Provide details of the hydrostatic tests performed on the BU-7 package. Include the package configuration, test setup, and package closure method.

(b) State whether the packages were newly fabricated or were packages which had been in service. Justify that the tests are representative of packages which are at the end of their service life.

(c) State how many specimens of each package type (BU-7 and BU-J) were tested. Note that Appendix B of the application dated December 3, 1993, states that only one BU-7 specimen was tested.

(d) Describe how the pass/fail determination was made.

(e) State how many specimens of each package type failed the test.

(f) Explain how the tests conducted on the BU-J package are relevant to the BU-7 package, considering any differences in the design, the dimensions, or the materials of construction.

4. Figure No. 10 in Appendix B of the application is incorrectly labelled. It does not appear that this is a photograph of drum No. K-1878 (see, for example, Figure No. 11 in the same appendix). In Figure No. 10, the bolt which secures the drum locking ring appears to be broken. Provide a description of the damage sustained by this bolt. If possible, provide an additional photograph which clearly shows that the bolt did not break due to the 30-foot drop test.

Criticality

1. The structural analysis of the product pails (Attachment B of supplement dated December 14, 1993) is not sufficient to show that the pails can reliably confine uranium oxide powder. Note that Figure 37 of the application clearly shows damage to the closure and deformation of the lid of the 5-gallon product pails following the accident test sequence. Note also that there are no test results available for the 3-gallon product pails. Revise the criticality analyses to consider that the uranium oxide powder may be released from the product pails under accident conditions.

2. Describe the method for benchmarking GEMER and identify the critical experiments used. Show that the biases presented in the application (including a bias of zero in cases where the code over-predicts k_{eff}) are proper and conservative for each of the H/U-235 ratios.

Operating Procedures

Specify the steps that will be taken before each shipment to verify that the product pails and inner container have been properly closed. Include a leak test to demonstrate that each inner container, as assembled for shipment, is water-tight. Specify the test method, the maximum acceptable leak rate, and the sensitivity of the leak test.

Acceptance Tests

1. Describe the method used to leak test each inner container before its first use. Specify the sensitivity of the leak test and the criteria for accepting the inner container. Include a sketch of the test set-up. Note that the leak test should be performed on the containment system as assembled for shipment, that is, all components of the containment system (drum, lid, and gasket) should be the components actually used for shipment. Also, the leakage flow direction during testing should be the same as in operation, i.e., into the inner container. Test methods using flow in the reverse direction should be justified.

2. The criticality analysis considers the presence of boron in the phenolic foam insulation. Revise the acceptance tests to include verification that boron is present and evenly distributed within the foam. State the criteria for accepting the foam.

Maintenance Program

1. Revise the maintenance program to include procedures for ensuring the reliable performance of the inner container as a water-tight containment system throughout its entire service life. These procedures should be performed annually and should include:

    a. A leak test which verifies that the inner container remains water-tight.

    b. Verification that the inner container welds, inner surface, and outer surface are free of corrosion, cracks, and other damage which could compromise the water-tightness of the package.

2. Revise the maintenance program to include annual inspection of the phenolic foam insulation. The annual inspection should include verification that the foam has not retained moisture, that the foam has not deteriorated, and that the boron content is within acceptable limits.

Drawings

Provide drawings of the 3- and 5-gallon pails. Include the following information on the drawings: dimensions, tolerances, material specifications, applicable codes and standards for fabricating and acceptance testing the pails, and details of the pail closure.

GE Nuclear Energy

Nuclear Fuel & Components Manufacturing
General Electric Company
P.O. Box 780, Wilmington, NC 28402
919 675-5000

March 18, 1994

Mr. Cass R. Chappell, Section Leader
Cask Certification Section
Storage and Transportation Systems Branch
Division of Industrial and Medical
Nuclear Safety, NMSS
United States Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: Response to Letter Dated February 17, 1994

Reference: Docket 71-9019

Dear Mr. Chappell:

This responds to your letter dated February 17, 1994, which requested additional information regarding GE's Consolidated Application dated December 3, 1993, seeking a revision to Certificate of Compliance USA/9019/AF for the Model No. BU-7 shipping container.

The letter also requested additional information concerning two subsequent submittals dated December 14 and 22, 1993, provided by GE at the NRC's request. In this regard, GE emphasizes that the additional submittals dated December 14 and 22, 1993, were not provided to satisfy any applicable regulatory requirement. Rather, GE voluntarily submitted that information solely to assist the NRC in its review of the Consolidated Application.

We have also noted that certain questions, including requests for justification, ask for information beyond that required in the NRC regulations. Nevertheless, GE has sought to cooperate by providing responses to each question. Although, in some cases, the information provided may not be in the level of detail requested, GE believes sufficient information has been submitted to demonstrate compliance with applicable NRC requirements.

Attachment 1 to this letter contains responses to the questions contained in the NRC letter dated February 17, 1994. Attachment 2 to this letter contains page changes to the December 3, 1993, Consolidated

Application reflecting any revisions needed on the basis of additional information contained in Attachment 1.

As we have previously indicated, GE strongly believes that the BU-7 package meets all applicable NRC regulatory requirements, and that the design and test data contained in the December 3, 1993, Consolidated Application clearly support this conclusion. Indeed, based upon substantially the same information as in the December 3, 1993, Consolidated Application, and as recent as 1988, the NRC concluded, in its Safety Evaluation Report, that the BU-7 package "meets the performance requirements of 10 CFR Part 71", and issued an approval.

Given this recent prior approval, GE's submittal of additional, detailed and sound engineering analyses, results of tests performed on the Model BU-7 and Model BU-J shipping containers to satisfy Japanese registration requirements, and the fact that the packaging design has remained essentially unchanged since 1974, GE knows of no reason why similar approval of the pending application should not be given for the BU-7 container at this time. The imposition during the current review of any requirements beyond those set forth in the regulations would improperly deprive GE of the ability to use shipping containers that have repeatedly been found to comply with the safety requirements of the NRC and foreign authorities.

As you know, the NRC first restricted the use of moderation exclusion and pail integrity for materials enriched above 4% U-235 in June, 1992. Because of the need to continue shipping, GE agreed to these highly conservative restrictive limitations as a temporary and expedient solution. Subsequently, the NRC also required GE to apply these same restrictive limitations to all enrichments 4% U-235 and below in the then upcoming certificate renewal. GE again complied with these restrictive limitations and was forced to purchase and lease hundreds of BU-J containers. These actions have cost, and continue to cost GE over a million dollars and severely affect our ability to participate in the international market.

Approval of the Consolidated Application would remove restrictions that are unwarranted under applicable regulatory requirements, and would be consistent with prior NRC approvals that were based on the same such requirements. In light of the substantial economic hardship imposed on GE by the current restrictions on transport in the BU-7 package, GE requests that the NRC complete the review of this current Consolidated Application as soon as possible.

Ten copies of this response are being provided for use during the review. Should the NRC have any questions regarding this matter, please contact me at (910) 675-5656.

Sincerely,

GE NUCLEAR ENERGY

C. M. Vaughan

C. M. Vaughan, Manager
Regulatory and EHS

Enclosures

cc: Mr. Charles J. Haughney (w/enclosure)
Mr. Carl J. Paperiello (w/o enclsoure)

ATTACHMENT 1

Structural

1. **The buckling analysis of the inner container (supplement dated December 14, 1993) shows a margin of only 15% against buckling under 21 psi external pressure. Section VIII of the ASME Code would require a thicker shell to resist 21 psi external pressure, and Section III of the Code would require a safety margin greater than 15% against buckling. The ASME Code applies to quality vessels, whereas the BU-7 inner container is not designed, fabricated, or inspected to the same standards as vessels which meet the ASME Code. Further, the buckling analysis does not account for possible deterioration of the container during service (note that most of the packages in use are at least 10 years old). Justify that the BU-7 containment system has an adequate margin of safety against buckling. Specify the code or standard used for design of the containment vessel of the BU-7 package. Show that this code or standard allows a margin of safety as small as 15% against buckling, and justify that this code or standard is appropriate to use for the containment system in the Model BU-7 package. Note that the integrity of the containment system is relied upon to ensure criticality safety under accident conditions.**

Section 71.31 requires that an application contain (1) a package description as required by § 71.33; (2) a package evaluation as required by § 71.35; (3) a Quality Assurance (QA) program description as required by § 71.37; and (4) an identification of the proposed fissile class.

The package description required by § 71.33 is provided in Section 2 of the BU-7 Shipping Package Consolidated Application (December 3, 1993).

With respect to package evaluation, § 71.35(a) requires a demonstration that the package satisfies the standards specified in Subparts E and F. Nothing in those regulations requires that an applicant provide structural design calculations. Instead, it is clear that the applicant is simply required to show that his package satisfies the tests specified in the regulations. For example, § 71.43(f) requires that a package "be designed, constructed and prepared for shipment so that under the tests specified in § 71.71 (Normal Conditions of Transport) there would be no loss or dispersion of radioactive contents, no significant increase in external radiation levels, and no substantial reduction in the effectiveness of the packaging." The key regulatory requirement is § 71.41, "Demonstration of Compliance," which specifically states "(a) The effects on a package of the tests specified in § 71.71 (Normal Conditions of Transport) and the tests specified in § 71.73 (Hypothetical Accident Conditions) must be evaluated by subjecting a sample package or scale model to test...." (emphasis added). This is precisely the package evaluation contained in Section 3 of the Consolidated Application.

With regard to § 71.73, GE maintains a quality assurance program for radioactive material shipping packages as approved under Docket 71-0254 dated October 5, 1989.

Thus, the Consolidated Application contains all of the information regarding structural adequacy of the BU-7 package needed to satisfy NRC regulatory requirements. Similar information was found fully acceptable by the NRC in its initial approval of the BU-7 package on August 6, 1974, and in subsequent approvals, including the Safety Evaluation Report issued on March 23, 1988.

The buckling analysis submitted by GE on December 14, 1993, was not provided to satisfy any applicable regulatory requirement. NRC reviewers had questioned whether the package was sufficiently robust to be leak tight so that moderation exclusion could be assumed in the criticality analysis. For regulatory purposes demonstration of such robustness was conclusively made through compliance with the tests specified in 10 CFR Part 71. However, since GE had performed a buckling analysis that was supportive of the acceptable test results and observed performance to date, GE was pleased to volunteer that information to the NRC, as we also volunteered the test data on 424 BU-7 18-gauge inner containers tested in 1982/83.

It should be further emphasized that the regulations do not require that a transportation package be designed or fabricated to ASME Code standards for vessels or to any other specified code or standard. Nor do they require that a buckling analysis be performed in accordance with the ASME Code standards for vessels or to any other code or standard. The ASME Code is intended for pressure retaining components and does not apply to transportation packages. It is thus inappropriate to evaluate the results of the buckling analysis against ASME Code standards or any other code or standard.

The buckling analysis for the 18-gauge inner container provided in Attachment A of our December 14, 1993, letter was performed using ANSYS which is a nuclear industry code used in nuclear power plant structural buckling evaluation. The ANSYS finite element analysis is accepted by the NRC for reactor related calculations and by ASME. Our calculations were based on Newton-Raphson techniques and employed the STIF63 shell elements for the 18-gauge inner container buckling model. The calculations are rigorous, yet conservative, and well base-lined in the nuclear industry.

The regulations require the package to withstand a water pressure equivalent to immersion under a head of water of at least 50 feet (21 psig) for not less than 8 hours. The buckling analysis demonstrates that there is a minimum margin of safety of 15% against buckling of the 18-gauge inner container over and above the regulatory requirement. In view of the regulatory requirements and of the rigorous and conservative nature of the analysis, as shown above, these results provide additional support for the evaluation in the application (based on the required tests) that the package satisfies the requirements of the regulations.

With particular regard to leak-tightness of the inner containers of the BU-7 package, as shown in Table 3-2(5) of the Consolidated Application, GE satisfied the NRC regulatory requirements of an immersion test of a single container [10 CFR 71.73(c)(5)]. Not only did the buckling analysis demonstrate that the inner container would satisfy such test, but additional supporting data is available in the information submitted in Attachment C to our December 14, 1993, letter. This documents that in 1982/83, 424 18-gauge inner containers for BU-7 packages satisfactorily passed a hydrostatic test at 21.4 psig as part of GE's program to register the BU-7 packages in Japan. Such additional data, although not required by NRC regulations, provides additional confidence in the leak-tightness of the BU-7 packages.

Question 1 includes a comment that the buckling analysis does not account for possible deterioration of the container during service. Just as the structural adequacy tests required under 10 CFR Part 71 are performed with new containers, the buckling analysis was performed using characteristics of a new container. The regulations do not require that the structural adequacy tests be performed again during the service life of the containers. It is apparent that the regulations rely upon the conservative structural adequacy demonstrated by satisfying the required tests before initial use, together with appropriate maintenance procedures and visual inspection before each use, to assure that the package will remain structurally adequate throughout its service life. Since the buckling analysis was intended to support the fact that the necessary structural adequacy existed before initial use, attempting to account for theoretical deterioration during service life would not be warranted or appropriate.

<u>Structural</u>

2. **For the 30-foot drop test, the BU-7 package was dropped on its top closure ring at approximately 45°. The closure ring was deformed on impact, and there was a slight opening of the drum lid. The subsequent puncture test was performed such that the package lid impacted the pin at a location away from the damaged area. The puncture test does not appear to have been performed in the orientation which would cause maximum damage to the package closure. The performance of the containment system (i.e., the ability of the inner container to exclude water) depends on the condition of the gasket after the fire test. The condition of the gasket after the fire test depends on the drum remaining closed. (Note that the insulating foam is charred all the way to the gasket after the fire test, as shown in Figures 35 and 36 of Appendix B of the application.) Justify that the 30-foot drop and puncture tests were performed in the most damaging orientation with respect to maximizing damage to the closure from the puncture test, and subsequently to the gasket from the fire test. Alternatively, perform additional 30-foot drop, puncture, and fire tests of the BU-7 package. The 30-foot drop and puncture tests should be performed in the orientation which produces maximum cumulative damage to the package closure.**

With regard to the Hypothetical Accident Condition tests in the BU-7 Test Report dated April 25, 1980, and submitted as required by regulation, two containers were subjected to the free drop and puncture tests (K-1878 and K-0174). The GE testing engineer decided to perform tests on two BU-7 packages (rather than testing a single package, as permitted by the regulations) in order to be able to perform the drop-test at differing orientations, thus providing additional assurance of testing package performance with maximum damage.

GE has re-evaluated the test information and interviewed the engineer who conducted the tests, and continues to believe that the tests were performed properly and to the requirements of § 71.32(a) and (c)(1) & (2) including the concept of cumulative damage.

Question 2 seems to be focused on whether the puncture test on package No. K-1878 was performed in the most damaging orientation, taking into account the damage to that package resulting from the 30-foot drop test. The following explanation for the selected orientation was developed using information obtained from the test report, test notes and engineers including the test engineer.

As the NRC is aware, the puncture test consists of a 40-inch (1 meter) drop onto a steel bar which does not produce a large amount of energy (1,233 ft-lbs versus 11,098 ft-lbs in the 30-foot drop). Therefore, given the design of the container, virtually all of the energy must be absorbed by the container to produce the maximum damage. The orientation, therefore, must provide for a direct impact rather than a glancing blow which dissipates the energy. Most of the surfaces and features of the BU-7 package are such that impacts on them would produce glancing types of collisions and the energy would not be transferred in any appreciable quantity to the container.

The test engineer was consulted to determine the decision process used to select the location for the penetration test. The engineer recalled and produced notes from preliminary testing work done March 10-13, 1980, describing four BU-7 containers which were challenged by several tests including the puncture test using both a sharp cornered and a rounded corner impact spike. Strikes from several different angles and to several different surfaces, including the closure ring and bolt, were completed. Based on all this information, the only type of strike which produced increased damage to the package were those strikes to the thin flat surfaces.

The preliminary test data as well as the damage to package No. K-1878 from the 30-foot drop test (Figures 11-14, Appendix B, Consolidated Application dated December 3, 1993) was considered in selecting the orientation and location of the puncture test on that package. While there was a slight opening of the cover where the closure ring of No. K-1878 was deformed, it should be noted that the closure and closure ring showed no sign of near failure. Thus, orienting the puncture test for No. K-1878 so that the deformed portion of the closure ring would hit the steel bar at an angle would not have absorbed as much energy and thus would not have been expected to cause as much damage to the container as was the case observed in preliminary tests.

In selecting a flat surface location for performance of the puncture test, it should be noted that, at the location of the deformed closure ring, the flat drum top surface is gently warped in such a manner that it would serve as an impact limiter in the collision, thereby reducing the ultimate damage. There was no crimping or metal tear or stretching that would indicate a special weakness in the already damaged location. Accordingly, the engineer selected a location away from this region because data from the preliminary tests indicated that the maximum damage would be produced by a strike on the flat thin surface and that such maximum damage was appropriate in evaluating the cumulative effect.

In evaluating the effects of the puncture tests (Figures 18 and 19, Appendix B, Consolidated Application dated December 3, 1993) the engineer found a slight indentation of only about 1/4" depression in both containers. This represents very minor damage and, as can be seen, there is no indication that the puncture test produced conditions that degraded the container's ability to withstand the sequence of tests. With the minor amount of energy involved, as evidenced in the photographs, it is reasonable to conclude that more serious damage would not have been done by dropping the container in other orientations or locations as suggested and the preliminary test data serve to reinforce this fact.

The NRC has not found error with the test during past reviews. In the March 23, 1988, Safety Evaluation Report, Section D, the NRC specifically addressed consideration of the cumulative effects of the hypothetical accident conditions and found the tests and results acceptable. In addition, it should be noted that, in this Safety Evaluation Report the NRC stated (at p. 5) that the "staff compared the results of similar shipping packages drop tested 30-feet in different orientations with the Model No. BU-7 package and concluded that different drop orientations would not result in the loss of structural integrity for the Model No. BU-7 package."

General Electric strongly believes that the test results that have been found adequate in the past continue to be adequate and that no further testing or justification pursuant to § 71.71 or § 71.73 is required.

Structural

3. The application (supplements dated December 14 and 22, 1993) discusses hydrostatic tests that were performed on BU-7 and BU-J packages. The application is not clear with respect to the details of the tests. Revise the application to clearly address the following:

(a) Provide details of the hydrostatic tests performed on the BU-7 package. Include the package configuration, test setup, and package closure method.

(b) State whether the packages were newly fabricated or were packages which had been in service. Justify that the tests are representative of packages which are at the end of their service life.

(c) State how many specimens of each package type (BU-7 and BU-J) were tested. Note that Appendix B of the application dated December 3, 1993, states that only one BU-7 specimen was tested.

(d) Describe how the pass/fail demonstration was made.

(e) **State how many specimens of each package type failed the test.**

(f) **Explain how the tests conducted on the BU-J package are relevant to the BU-7 package, considering any differences in the design, the dimensions, or the materials of construction.**

(a) The BU-7 hydrostatic tests discussed in GE's December 14 and 22, 1993, submittals were performed to satisfy Japanese registration needs, not NRC requirements. They were provided to the NRC as further substantial evidence of the 18-gauge inner container's leak tight design.

Figure I attached defines the test set-up showing air supply valves, pressure gauge, water supply location, vent, configuration, etc. The package component tested in each case was a newly constructed 18-gauge inner container fitted with the specified 3/16" thick, 1-1/2" wide steel flange. Closure of the 3/16" thick steel lid was accomplished using a gasket and twelve 5/16-18 carbon steel bolts as specified for the package. The tests were performed on 18-gauge inner containers at a minimum pressure of 21.4 psig for 8 hours. In passing the test, there was no visual water in-leakage.

In the same submittals GE provided information on immersion tests on BU-Js performed by the Saito Company for Japanese Nuclear Fuel (JNF) in Japan to meet Japanese registration requirements. Figure II attached summarizes the Japanese test set-up that JNF reported to GE. The tests are identical to the tests performed for the BU-7s except that the Japanese test set-up accommodates up to eight containers per test. Japan performs 100% leak tightness tests for all BU-Js.

GE does not believe that it is appropriate or necessary to incorporate data generated in satisfying Japanese registration requirements in GE's application for NRC certification since regulations do not require such data. This test data does, however, provide supplementary support for GE's claim of leak tightness for, and the NRC's approvals of, the BU-7's leak

FIGURE I

BU-7 TEST CONFIGURATION

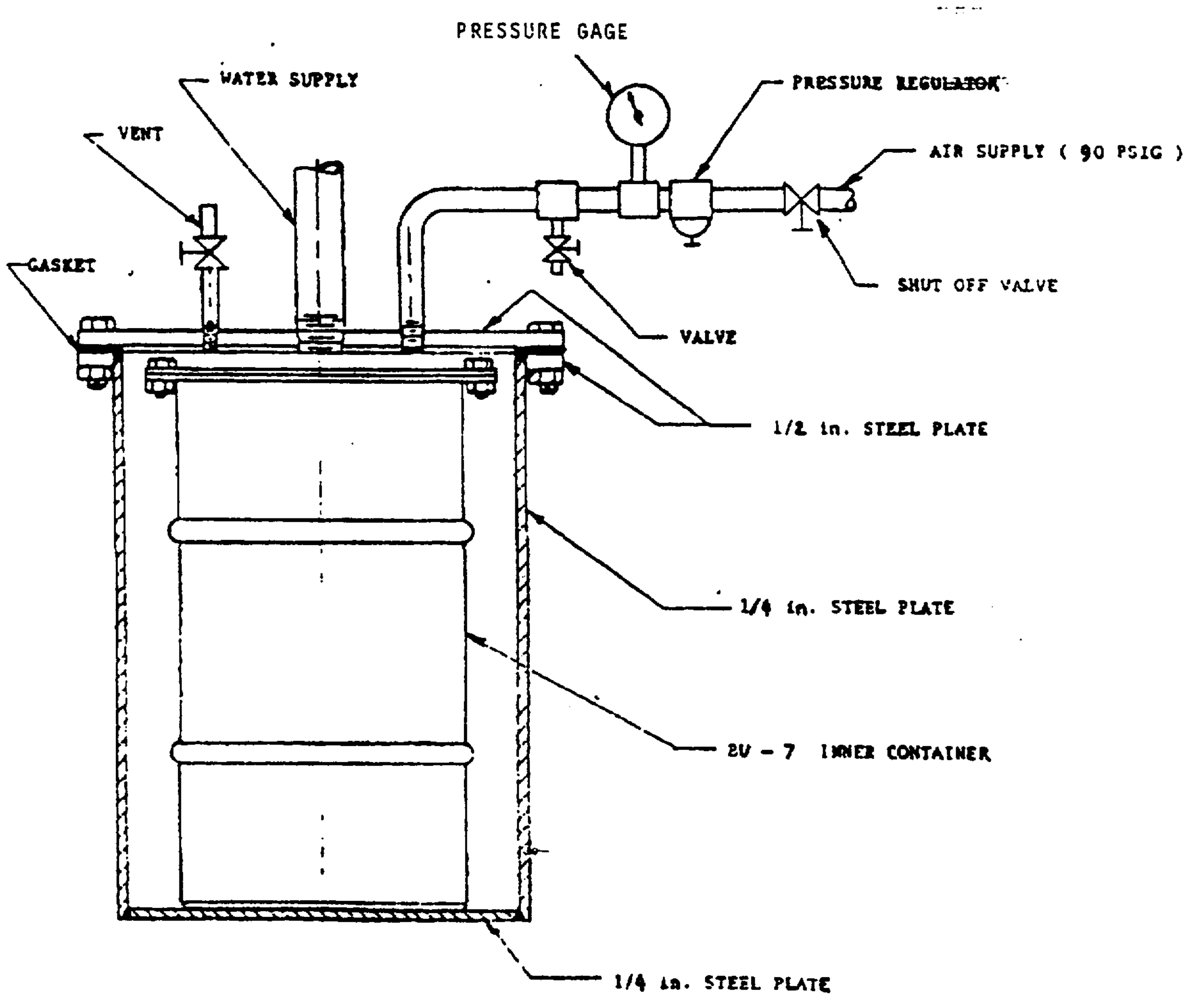
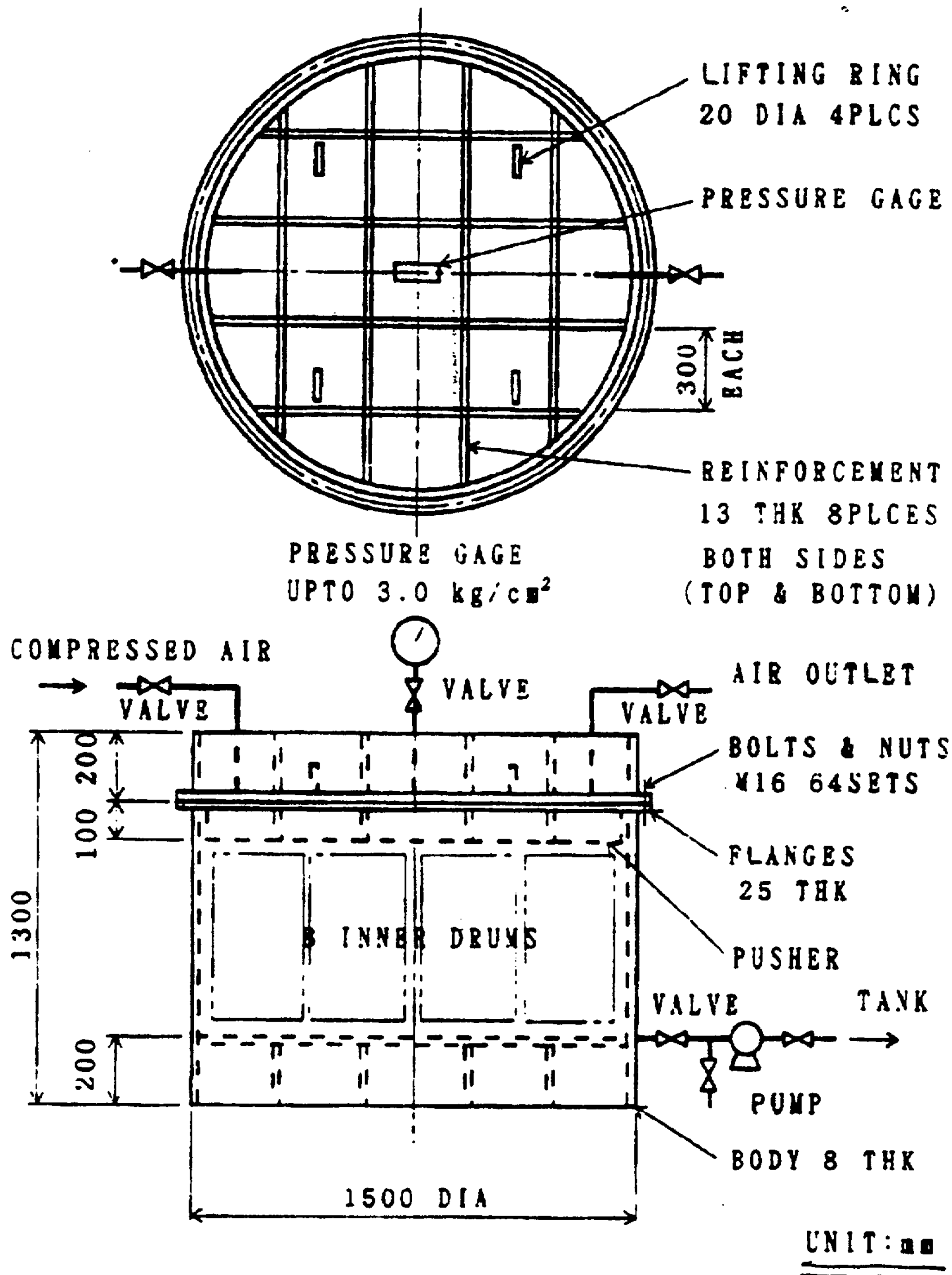


FIGURE II

BU-J TEST CONFIGURATION



TANK for BU-J Pressure Test

tightness as stated in the SERs attached to the Certificate of Compliance.

(b) The hydrostatic tests discussed in GE's December 14 and 22, 1993, submittals were performed to satisfy Japanese registration needs, not NRC requirements.

Notwithstanding the reason for the testing, all tests were performed on new 18-gauge inner containers as specified in § 71.73(c)(5).

As described in the response to Question 3(a), the hydrostatic tests performed for purposes of Japanese registration were the functional equivalent of the immersion tests performed for purposes of NRC certification under § 71.73(c)(5). That NRC regulation explicitly requires that the immersion test be performed with an undamaged container and does not require that justification be provided that the test is representative of packages which are at the end of their service life. It is apparent that the regulations rely upon the conservative leak tightness demonstrated by satisfying the required test before initial use, together with appropriate maintenance procedures and visual inspection before each use, to assure that the inner container will remain leak tight throughout its service life. Thus, the hydrostatic tests performed on new BU-7 and BU-J containers for purposes of Japanese registration requirements provide additional support for confidence in the leak tightness of BU-7 containers throughout their service life, just as the immersion test performed on new containers under § 71.73(c)(5) provide such confidence, when coupled with appropriate maintenance procedures and visual inspection. The maintenance and visual inspections for the BU-7 packages have been performed in accordance with GE programs approved by the NRC. See March 23, 1988, Safety Evaluation Report, pp. 8-9.

(c) Appendix B of the December 3, 1993, application addresses the BU-7 tests that were performed to satisfy NRC certification requirements for BU-7 packages in accordance with the applicable provisions of 10 CFR Part 71 in effect at that time. The number, identity, and tests are summarized in Section 2.1 and detailed within the report.

GE's submittals of December 14 and 22, 1993, provided BU-J test information and the BU-7 test data for registration in Japan. This information was provided over and above any NRC requirements to assist in the NRC's evaluation and general demonstration of the leak-tight nature of the container design.

424 BU-7s were satisfactorily tested as part of GE's registration of the BU-7 package in Japan. Japanese authorities accepted these test results as sufficient for registration of these BU-7 packages.

GE also has records of satisfactory hydrostatic test results for 1280 BU-J packages. GE recently purchased 431 BU-J packages that were tested as part of this group. GE does not know the total number of BU-J packages that have been satisfactorily tested in Japan, but it has been reported that every BU-J package is tested before registration can be obtained.

The results of these Japanese tests on large numbers of BU-7 and BU-J containers serve to reinforce the validity of GE's design calculations and test results.

The tests performed to meet 10 CFR Part 71 requirements for NRC certification of the BU-7 package are described in Appendix B of the December 3, 1993, application. Attached to Appendix B is Appendix 3 "Test Report BU-5 and BU-7 Container Pressure Test" dated February 10, 1978, which discusses the 50-foot immersion test. A visual examination for moisture was used.

(d) The hydrostatic tests discussed in GE's December 14 and 22, 1993, submittals were performed to satisfy Japanese registration needs, not NRC requirements. A visual examination for moisture was used and failure was defined as a visual indication that water was present in the inner container.

(e) The objective of the tests described in GE's December 14 and 22, 1993, submittals was to verify that inner containers did not leak and then pass them on to fabrication. As a result, GE did not establish a procedure to record any leaking BU-7 inner containers and does not have any documentation of leaking containers. Similarly, GE's records of tests of BU-J containers does not mention any failures. Questioning those involved with the tests of the BU-7 containers bring no recollection of failures.

(f) The BU-7 hydrostatic test results discussed in GE's December 14 and 22, 1993, submittals were performed to satisfy Japanese requirements which are similar to those found in § 71.73(c)(5) dealing with immersion to the equivalent of 50 feet in water.

The BU-7 and BU-J 18-gauge inner containers are nearly identical as can be seen in Figure III attached. Both the BU-7 and BU-J use an 18-gauge minimum steel drum with two rolling hoops. The drum dimensions are near identical. The closure flange and lid are identical except for the size of the bolt closure holes and weld nuts. The closure bolts for the BU-7 are 5/16-18 (.313") carbon steel and for the BU-J are 8 mm (.315"). The BU-7 uses a silicone gasket with detailed specifications to assure its quality. The BU-J specifies only 3mm butyl rubber. The BU-7 closure procedure specifies prescribed torque requirements whereas the BU-J is silent on this point.

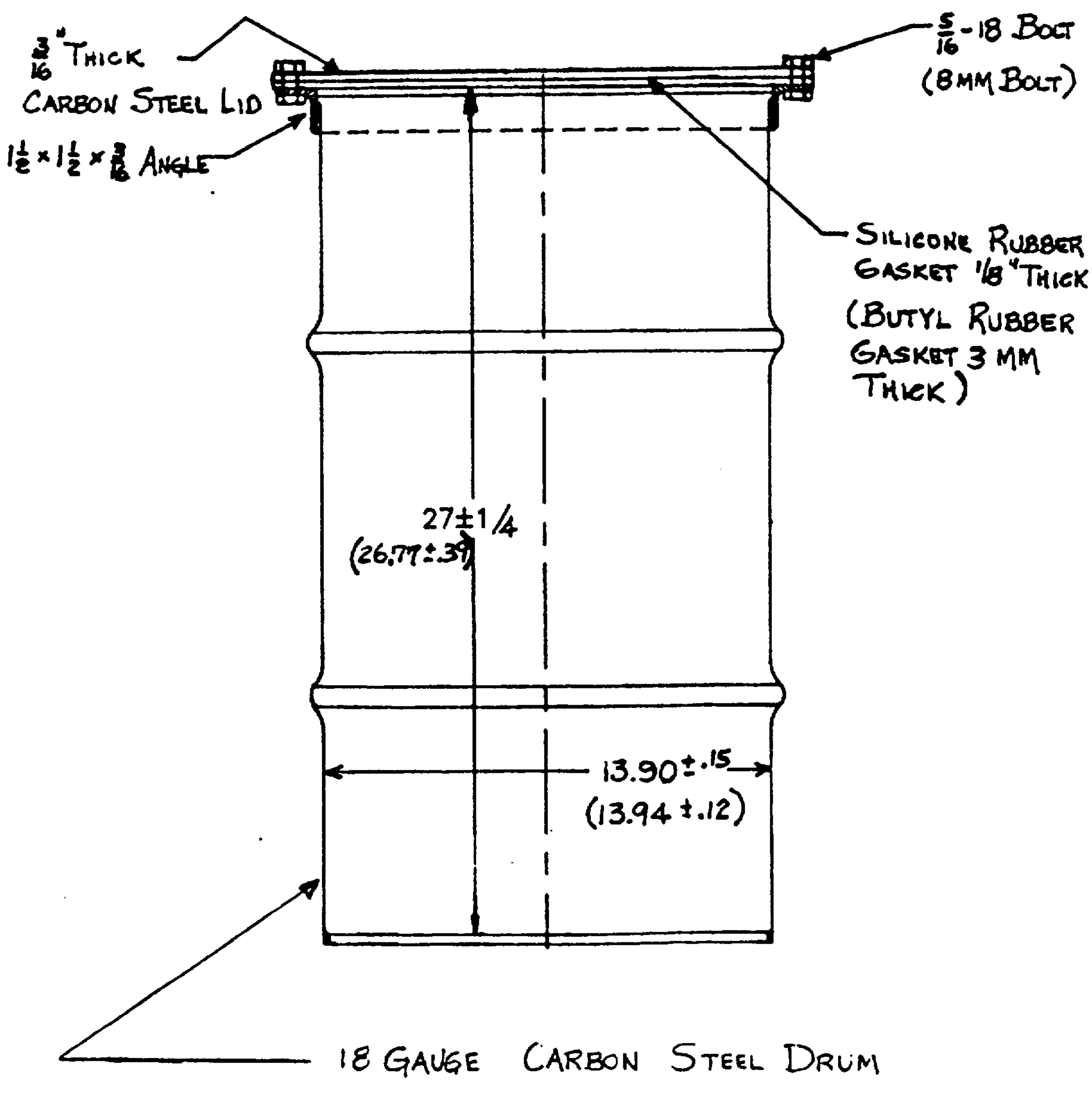
Since the BU-7's design, dimensions, and materials of construction of the 18-gauge inner container are equal, or in some cases, superior (i.e., gasket and closure torque), and the tests are performed to the same specification, it is clear that the BU-J testing is mutually supportive in the area of leak tightness of the 18-gauge inner container.

Structural

4. Figure No. 10 in Appendix B of the application is incorrectly labelled. It does not appear that this is a photograph of drum No. K-1878 (see, for example, Figure No. 11 in the same appendix). In Figure No. 10, the bolt which secures the drum locking ring appears to be broken. Provide a description of the damage sustained by this bolt. If possible, provide an additional photograph which clearly shows that the bolt did not break due to the 30-foot drop test.

FIGURE III

BU-7/BU-J INNER CONTAINER COMPARISON



NOTE: BU-J DIMENSIONS, IF DIFFERENT FROM BU-7, ARE SHOWN IN ( ).

GE has reviewed the report and it appears that Figure 10 is incorrectly labeled. Figure 10 more closely represents the damage done to the top of K-0174 (see Figure 18) and the report has been modified accordingly. Figures 11, 12, 13, and 14 allow easy identification as K-1878.

Figure 10 shows no breakage or damage to the closure bolt and the test engineer interviewed recalls none. The darkened area in question is a shadow as can be seen from the lighting in the photograph (cf the point of the closure ring at the nut location). In support of this position, GE has located two additional photographs from the photo sequence of the bolt on container K-0174 which more clearly show from two different perspectives that the bolt is not damaged. The photographs are enclosed and are being incorporated into the test report (Figures 10A and 10B) to avoid the ambiguity generated by Figure 10.

Criticality

1. **The structural analysis of the product pails (Attachment B of supplement dated December 14, 1993) is not sufficient to show that the pails can reliably confine uranium oxide powder. Note that Figure 37 of the application clearly shows damage to the closure and deformation of the lid of the 5-gallon product pails following the accident test sequence. Note also that there are no test results available for the 3-gallon product pails. Revise the criticality analyses to consider that the uranium oxide powder may be released from the product pails under accident conditions.**

The analysis presented in Appendix B of the supplement dated December 14, 1993, clearly demonstrates from an engineering standpoint that the product pail will contain the powder. The calculations included as Appendix B to our letter of December 14, 1993, CM Vaughan to CJ Haughney, were performed using first principles of engineering and standard formulas. GE knows of no deficiency in this analysis nor has the NRC identified any.

Notwithstanding the analysis, Figures 35, 36, 37, and 38 demonstrate the integrity of the 5-gallon product pails after the required test. There is no failure of the containment that would suggest a failure important to criticality safety under the terms of the safety analysis. Additionally, the test engineer recalls that, while there were some dents and deformation of the pails, the lids remained tight in all test cases.

The three-gallon pail is dimensionally identical to the five-gallon pail except that it is approximately 5 inches shorter than the five-gallon pail. Therefore, from a buckling standpoint, the three-gallon pail is stronger than the five-gallon pail since the maximum allowable stress is inversely proportional to pail height.

Criticality

2. Describe the method for benchmarking GEMER and identify the critical experiments used. Show that the biases presented in the application (including a bias of zero in cases where the code over-predicts k_{eff}) are proper and conservative for each of the H/U-235 ratios.

This information has already been supplied to the NRC Transportation Branch in a separate submittal dated May 23, 1990, related to the UNC-2901 package, NRC Certificate Number 6294.

With regard to bias, GE treats bias correctly. Where K_{eff} is underpredicted, the bias and the uncertainty are added to the result. Where K_{eff} is overpredicted, our current policy is not to apply a correction; which if applied, would in effect increase the maximum K_{eff} permitted in the analysis.

Operating Procedures

Specify the steps that will be taken for each shipment to verify that the product pails and inner container have been properly closed. Include a leak test to demonstrate that each inner container, as assembled for shipment, is water-tight. Specify the test method, the maximum acceptable leak rate, and the sensitivity of the leak test.

GE's program to ensure that package closure requirements are properly completed before shipment includes elements of the acceptance testing program, maintenance program, and operating procedures.

In order to assure that 18-gauge inner containers have been properly closed, the first requirement is that they satisfy NRC design, fabrication and associated acceptance testing requirements relating to leak-tightness. Under NRC regulations, this is demonstrated by subjecting a separate undamaged specimen to external water pressure of at least 21 psig for at least 8 hours in accordance with § 71.73(c)(5). As discussed in Table 3-2(5) of the Acceptance Tests portion of the Consolidated Application and as detailed in Appendix 3 to Appendix B of the application in a Test Report dated February 10, 1978, this immersion test was successfully performed. In addition, as discussed in the answer to Q-3, "Structural", GE has records showing that a substantial equivalent of the immersion test was also performed successfully with 424 BU-7 inner containers and at least 1,280 BU-J inner containers. Moreover, prior to first use of each 18-gauge inner container, GE has ascertained "that there are no cracks, pinholes, uncontrolled voids or other defects which would significantly reduce the effectiveness of the packaging..." in accordance with § 71.85(a). This included a submerged bubble pressure test at a minimum of 15 psig, as described in Section 5.2.1 of the Consolidated Application.

The product pails are not considered as a leak tight container as defined in § 71.73(c)(5) and § 71.85(a). Their function is to hold the authorized contents, UO_2 powder, etc.

The second set of requirements for proper closure is intended to assure that a properly designed and fabricated container has been appropriately maintained after initial use. This subject is covered by the Maintenance Program as described in Section 5.3 of the Consolidated Application dated December 3, 1993. The Maintenance Program for the 18-gauge inner container covers welded flange integrity, cleanliness and paint, gasket sealing surfaces, gasket inspection and replacement, bolt threads and holes. The Maintenance Program is not relevant to the product pails since they are a single trip container.

The third set of requirements for proper closure, is intended to assure that appropriate practices are used in closing the containers prior to each shipment. This subject is covered in the Operating Procedures described in Section 5.1, of our Consolidated Application dated December 3, 1993. Loading and closure is specifically addressed for the pails (5.1.1), the 18-gauge inner container (5.1.2) and final packaging and mechanical closure (5.1.5). All these elements cover configuration, container integrity, sealing surfaces, and closures, including torque and sealing requirements at key points. They fully satisfy the requirements of § 71.87(f) that prior to each shipment, the licensee determine that "the package has been loaded and closed in accordance with the written procedures."

GE believes this program to be well designed to ensure that containers are properly closed. The program has been highly effective in preventing improperly closed packages from leaving the site.

The same subject was evaluated by the NRC in the March 23, 1988, Safety Evaluation Report and found to be acceptable referencing Sections 5.1, 5.2, and 5.3, and including Condition 3 of the SER dealing with Acceptance Testing (5.2) and Maintenance Program (5.3).

There is no requirement in NRC regulations that each inner container be leak-tested for water-tightness before each shipment. As described above, the regulations require that a separate, undamaged specimen satisfy the immersion test of § 71.73(c)(5) and that all packages meet § 71.85(a) before first use. Not only is subjecting each inner container to a leak test for water tightness before each use not required, but it would be infeasible to do so for an assembled package. Subjecting such package to an immersion test would be the equivalent of destructive testing. Even though the inner container would not leak, the insulation between the inner container and the outer container would be ruined.

Acceptance Tests

1. **Describe the method used to leak test each inner container before its first use. Specify the sensitivity of the leak test and the criteria for accepting the inner container. Include a sketch of**

the test set-up. Note that the leak test should be performed on the containment system as assembled for shipment, that is, all components of the containment system (drum, lid, and gasket) should be the components actually used for shipment. Also, the leakage flow direction during testing should be the same as in operation, i.e., into the inner container. Test methods using flow in the reverse direction should be justified.

The method used to leak test the inner containers prior to first use is described in Section 5.2.1 of the Consolidated Application. This test involves submerging in water the inner container under 15 psig internal pressure for at least one minute. During this time the container is observed for visible bubbles to determine whether any leaks exist. A sketch of the test set-up is attached (Figure IV).

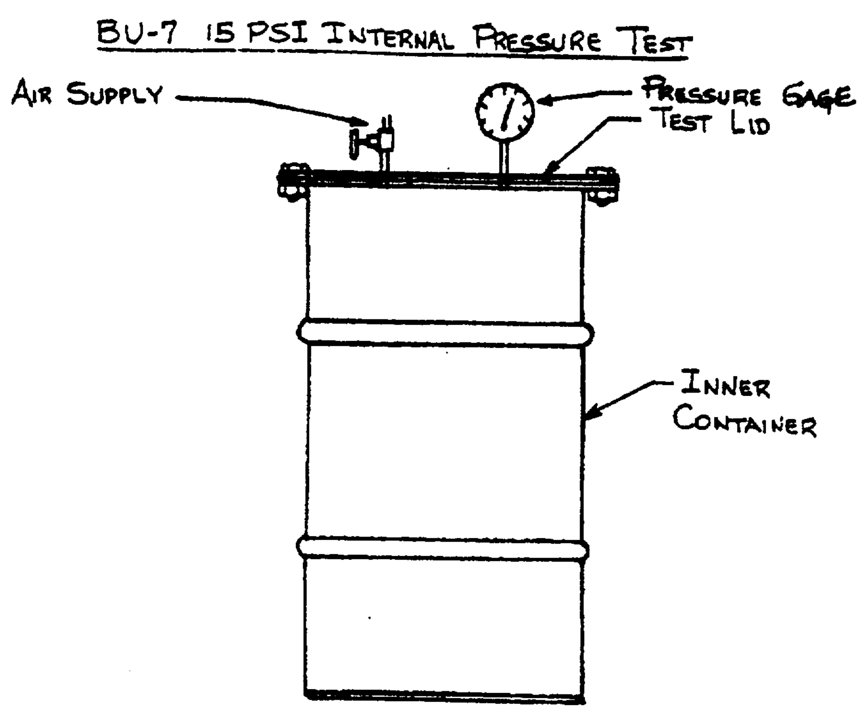
Leak testing prior to first use on each containment system as assembled for shipment (i.e., inner container, lid, gasket) is not required by the regulations. Neither do the regulations require consideration of, among other things, leakage flow direction when conducting the tests. § 71.85(a) provides that prior to first use of any packaging "[t]he licensee shall ascertain that there are no cracks, pinholes, uncontrolled voids, or other defects which could significantly reduce the effectiveness of the packaging...." The acceptance test performed by GE and set forth in Section 5.2 of the Consolidated Application clearly meets this requirement. In addition, the NRC found this testing acceptable (page 9) in its March 23, 1988, Safety Evaluation Report. Further, as noted above in GE's response to the NRC's question regarding "Operating Procedures," conducting such a leak test prior to shipment on existing "containment systems as assembled for shipment" would amount to a destructive test.

Acceptance Tests

2. **The criticality analysis considers the presence of boron in the phenolic foam insulation. Revise the acceptance tests to include verification that boron is present and evenly distributed within the foam. State the criteria for accepting the foam.**

GE has modified the acceptance criteria to include the requirement to verify the boron content in the foam. The acceptance criteria has been

FIGURE IV



TEST PROCEDURE:

1. COMPLETED INNER CONTAINER IS FITTED WITH A SILICONE RUBBER GASKET AND TEST LID BOLTED IN PLACE.
2. THE INNER CONTAINER IS PRESSURIZED TO 15 PSI WITH AIR AND THE VALVE CLOSED TO RETAIN THE PRESSURE.
3. THE INNER CONTAINER IS PLACED IN AN OPEN TANK OF WATER AND ROTATED TO EXAMINE THE GASKET AND FLANGE TO DRUM WELD, LONGITUDINAL WELD SEAM, AND BOTTOM ROLLED EDGE FOR LEAKS.
4. NO BUBBLES ARE ALLOWED FROM ANY EXTERIOR SURFACE.

established as ensuring a minimum value which exceeds the value in the criticality analysis by a minimum of 33% (i.e., this allows credit for no more than 75% of the boron verified as present in the foam). This acceptance shall only apply to packages fabricated after this approval is granted.

In 1993 the boron content of the foam was determined by destructive analysis for the current fleet of BU-7 containers. The sample plan incorporated three samples circumferentially at the top 1/3 (A1-3), mid 1/3 (B1-3), and lower 1/3 (C1-3) and a tenth sample from the bottom region. The results for the 29 BU-7s evaluated are reported in Table I (attached). The average value is 2.64 weight percent boron with a high value of 5.6% and a low value of 1.24%. This confirms that the boron content is typical of the specification used to make the foam even though it does not precisely match the value of 3.2% mentioned as nominal in specification (SP-9). The values are also far in excess of the 0.16 weight percent boron used in the criticality calculations to demonstrate safety.

Maintenance Program

1. **Revise the maintenance program to include procedures for ensuring the reliable performance of the inner container as a water-tight containment system throughout its entire service life. These procedures should be performed annually and should include:**

   a. **A leak test which verifies that the inner container remains water-tight.**

   b. **Verification that the inner container welds, inner surface, and outer surface are free of corrosion, cracks, and other damage which could compromise the water-tightness of the package.**

2. **Revise the maintenance program to include annual inspection of the phenolic foam insulation. The annual inspection should include verification that the foam has not retained moisture, that the foam has not deteriorated, and that the boron content is within acceptable limits.**

TABLE I

BU-7 Boron Sampling Results

(wt% B)

DRUM S/N	A1	A2	A3	B1	B2	B3	C1	C2	C3	4	Average	Std. Dev.	Count
258	2.83	2.37	2.30	2.29	2.48	2.79	2.45	2.69	2.55	2.53	2.53	0.19	10
361	2.66	2.63	2.57	2.79	2.58	2.67	1.80	2.37	2.39	2.47	2.49	0.28	10
374	3.20	2.81	2.76	2.89	2.73	2.79	2.66	3.12	2.45	2.59	2.80	0.23	10
484	3.41	3.41	3.11	2.29	3.46	2.76	2.19	2.36	2.21	2.55	2.78	0.53	10
553	2.53	2.45	2.45	2.73	2.39	2.16	2.32	1.66	2.73	2.81	2.42	0.34	10
672	2.78	2.03	2.21	2.55	2.86	2.03	2.46	2.69	2.07	2.08	2.38	0.33	10
1240	2.05	1.51	1.78	1.65	1.61	1.64	2.02	1.96	2.33	1.73	1.83	0.25	10
1585	1.99	1.79	2.27	1.99	2.11	4.25	1.63	1.54	1.54	2.51	2.16	0.80	10
1774	3.98	4.72	4.79	5.04	4.70	4.59	5.04	4.96	2.69	3.17	4.37	0.82	10
1951	2.28	2.04	1.98	1.75	1.97	2.15	2.82	1.83	1.79	1.52	2.01	0.36	10
1989	1.94	1.76	1.76	1.90	1.82	1.76	1.89	1.73	1.72	1.92	1.82	0.08	10
2911	2.45	2.29	2.50	2.40	2.15	2.25	2.50	2.51	2.55	2.63	2.42	0.15	10
3083	2.78	2.11	2.70	2.30	2.08	2.28	2.07	2.63	2.55	2.44	2.39	0.26	10
3151	2.76	2.32	2.25	2.67	2.72	2.69	2.68	2.51	2.60	2.20	2.54	0.21	10
3240	2.90	2.47	2.43	2.90	2.39	2.58	2.44	2.24	2.23	2.27	2.49	0.24	10
3483	1.31	2.58	2.21	1.64	2.75	2.09	2.05	2.37	2.99	3.13	2.31	0.58	10
6016	2.30	1.51	1.94	1.95	1.98	1.48	2.06	2.08	1.62	1.88	1.88	0.26	10
6190	1.38	1.38	1.33	1.54	1.41	1.24	2.15	2.22	2.13	1.91	1.67	0.39	10
6238	2.47	2.50	1.85	2.39	1.97	1.77	2.11	2.25	2.33	1.90	2.15	0.27	10
6573	2.02	2.23	1.64	3.08	3.06	3.56	2.01	2.89	1.77	2.59	2.49	0.65	10
6672	2.19	1.52	2.4	3.1	1.96	1.66	2.88	3.1	2.54	2.67	2.40	0.56	10
6733	3.55	3.65	5.6	5.36	3.67	4.29	2.23	2.57	2.76	3.57	3.73	1.11	10
6812	2.49	2.55	2.25	2.80	2.23	3.04	3.20	2.85	2.27	2.47	2.62	0.34	10
7101	3.84	3.34	3.28	3.29	3.69	4.16	3.95	3.83	3.91	1.80	3.51	0.67	10
7113	3.74	3.68	3.70	3.82	3.54	3.91	3.98	3.86	3.53	4.45	3.82	0.27	10
7334	3.31	3.31	3.44	2.28	3.25	2.96	3.2	3.1	2.90	4.10	3.19	0.46	10
7414	3.88	3.43	4.06	4.51	3.31	3.73	2.67	2.6	4.37	3.13	3.57	0.66	10
7546	1.26	2.52	3.57	3.04	1.63	2.39	3.62	2.93	3.29	3.12	2.74	0.79	10
7621	3.55	3.37	2.15	2.38	3.34	3.40	4.03	2.52	3.01	2.80	3.06	0.59	10
										Average	2.64		
										Std Dev	0.65		
										Count	29		

The applicable regulations do not require the suggested revisions to GE's existing Maintenance Program. Section 71.37(a) only requires the applicant to describe the quality assurance program for the maintenance of the proposed package. Pursuant to this requirement and as stated in Section 5.3 of the Consolidated Application, GE has implemented an effective maintenance program. Specifically, under GE's maintenance program inspections are performed before each use of the BU-7 outer drum and cover, the inner container and lid, and the boral liner, in accordance with detailed procedures to assure that the packages are in serviceable condition. The NRC approved this program in its March 23, 1988 SER and incorporated the boral liner November 19, 1993. In addition, GE submitted a quality assurance program, which the NRC similarly approved, on October 5, 1989.

GE inspects and tests BU-7 containers when new and before each use to assure that it is within the acceptable foam density bounds. If the weight of the container falls outside the accepted upper or lower density limits, that BU-7 is removed from service. This assures that the containers used do not contain excess moisture and that they do contain an adequate density of foam insulation.

In addition to the verification of the foam density, GE has verified the boron content of the foam in the current fleet after several years of service and for newly fabricated containers will verify the boron content during fabrication. Repeated boron verification is not necessary because there are no boron reduction scenarios that would not be detected by the inspection and maintenance program.

Drawings

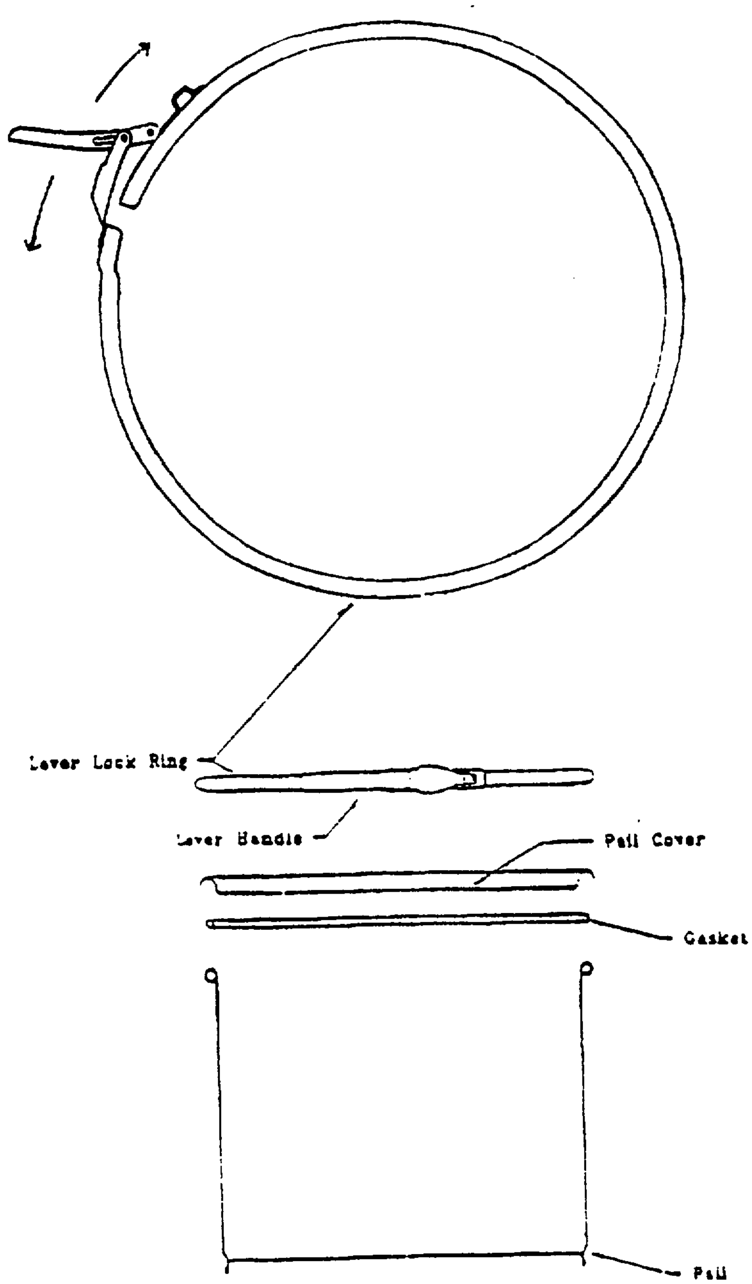
Provide drawings of the 3- and 5-gallon pails. Include the following information on the drawing: dimensions, tolerances, material specifications, applicable codes and standards for fabricating and acceptance testing the pails, and details of the pail closure.

GE purchases of the 3- and 5-gallon pails are specified to meet the ANSI MH-2-10-1979 document and in compliance with DOT regulations (pre-HM181) 49 CFR 178.131 for the DOT-37A80 container. Therefore, the drawings, dimensions, tolerances, material specifications, applicable codes and standards are contained in the above referenced ANSI standard and DOT regulation. The 24-gauge gasketed pail cover is held in place by a lever locking ring (see Figure V attached).

Acceptance testing of the pails is accomplished in accordance with 49 CFR 178.131-11 by the vendor.

FIGURE V

PRODUCT PAIL CLOSURE DEVICE



ATTACHMENT 2

The following page changes are for the BU-7 Consolidated Application dated 12/3/93.

1. Appendix B, Figure 10, has been changed to show the BU-7 serial number as K-0174 instead of K-1878. Asterisks in the right-hand margin show the location of the changes. Also added to the new Figure is the date of this transmittal (3/18/94) and that this is Revision 1 of Figure 10.

2. Figures 10A and 10B have been added. These are photos of BU-7 serial number K-0174 that have not been provided to the NRC in the past. They show that the bolt securing the outer ring was not broken. These two new figures are also identified with the drum serial number and the date of this transmittal.

3. Pages 21 through 24 of Section 5.2.1 of the 12/3/93 Consolidated Application have been modified to include the prior-to-first-use Acceptance Testing criteria for boron content in the foam. The dates on the changed pages (and those that changed as a result of pagination) have been changed to reflect the date of this transmittal. Also, the revision number has been changed on each page and asterisks are placed in the right-hand margin by the changes made to the text.

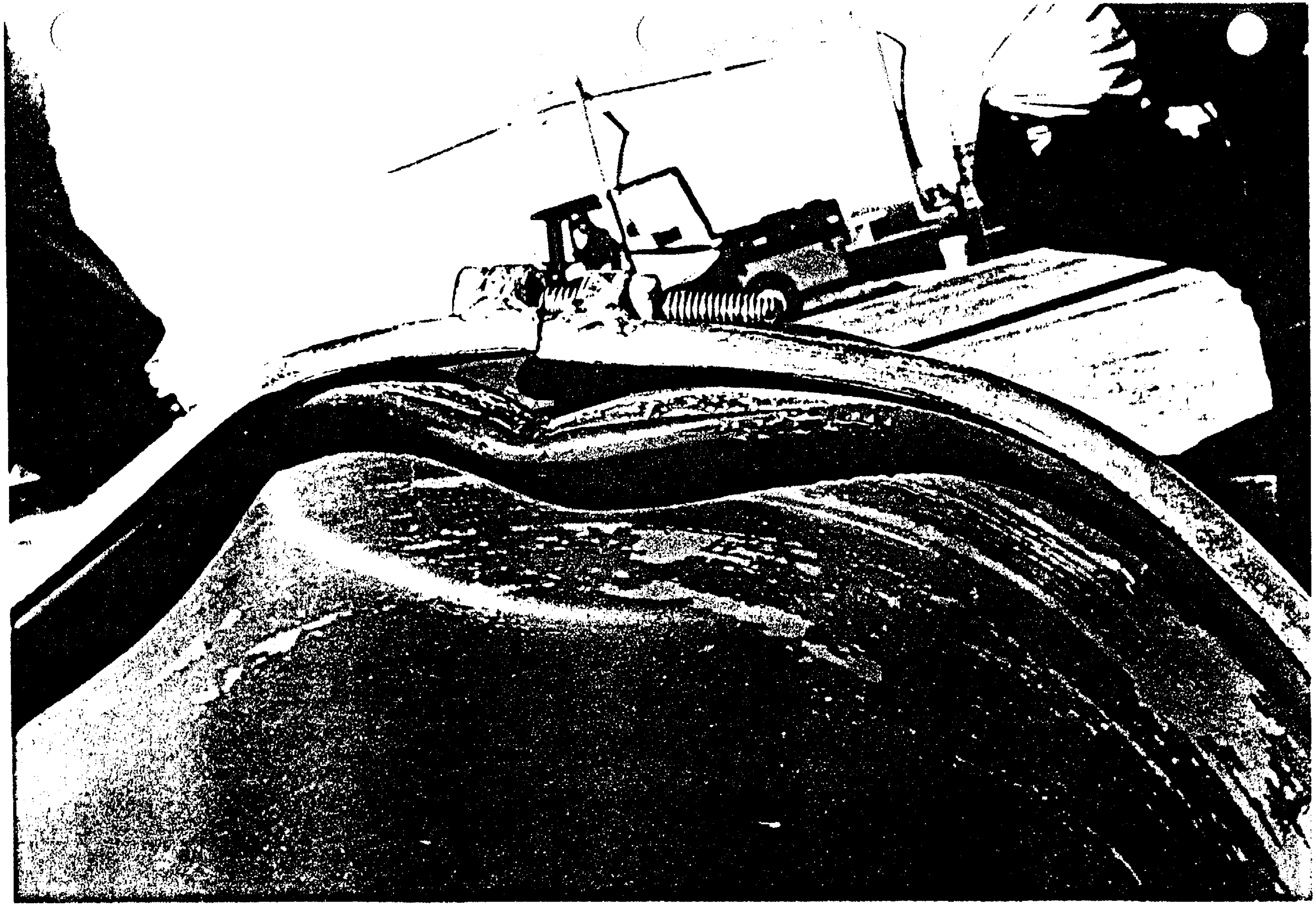


FIGURE 10 REV. 1 SUBMITTED 3/18/94

SERIAL NO. K-0174 AFTER IMPACT

* * * *

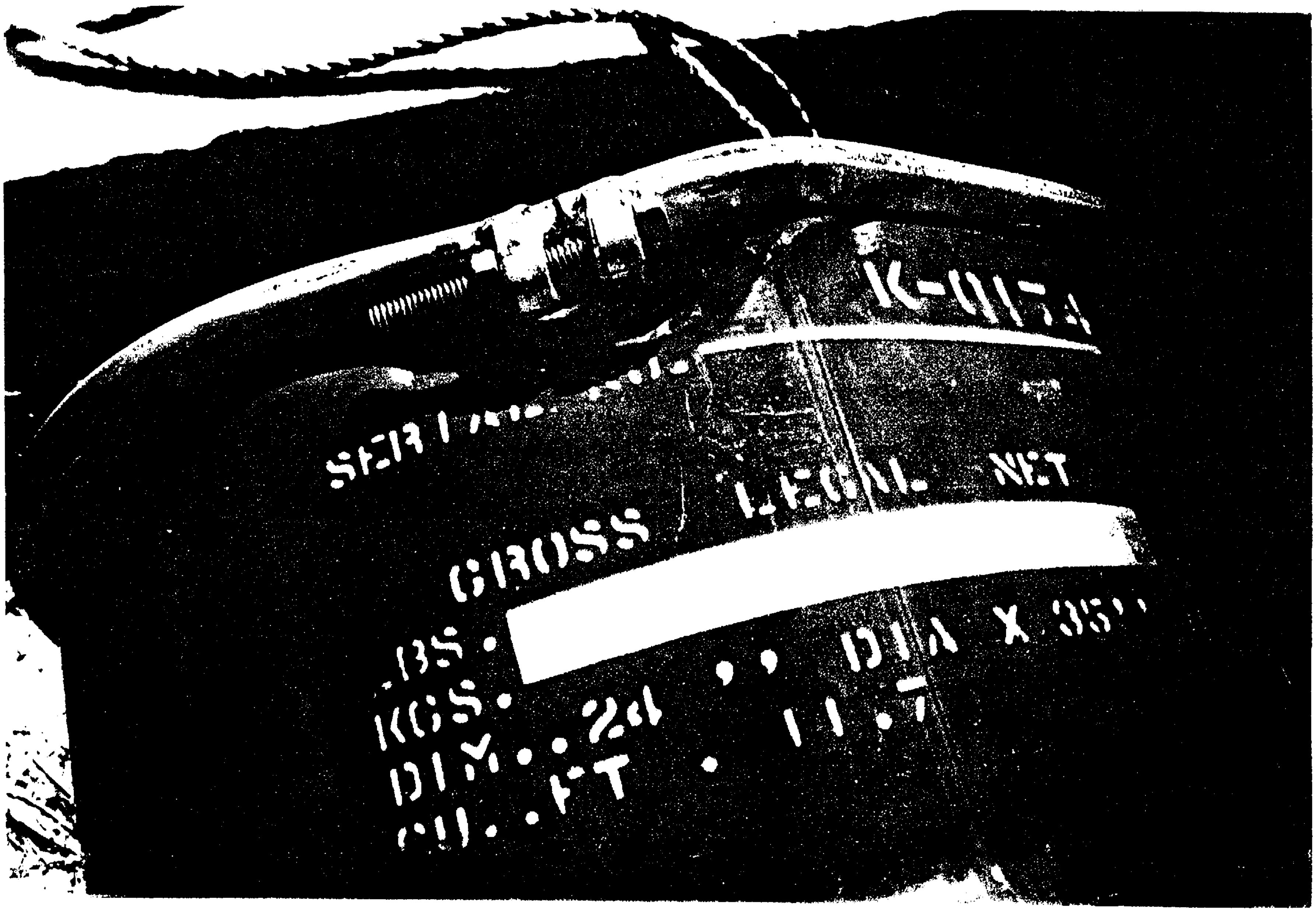


FIGURE 10A SUBMITTED 3/18/94

SERIAL NO. K-0174 AFTER IMPACT

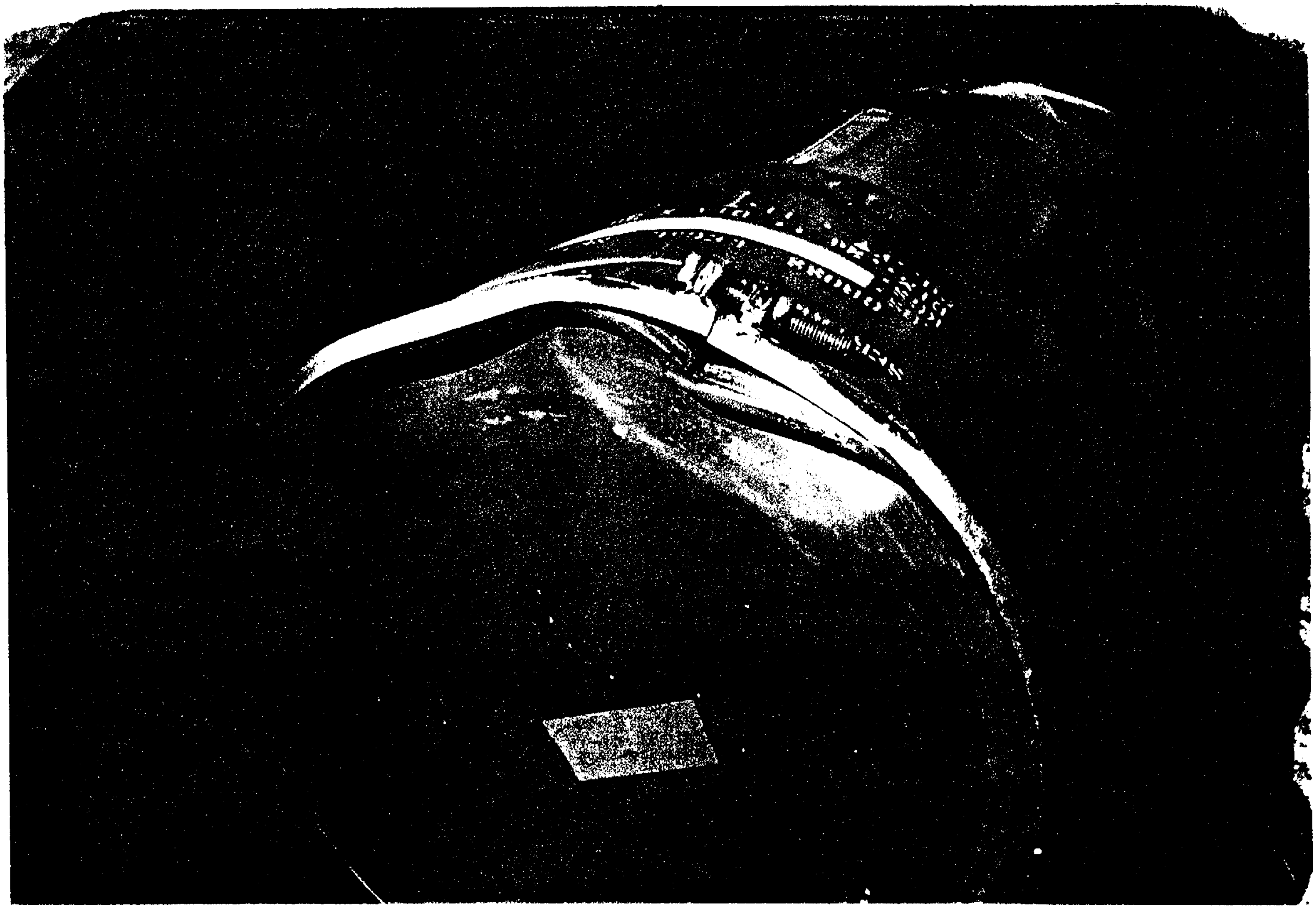


FIGURE 10B SUBMITTED 3/18/94

SERIAL NO. K-0174 AFTER IMPACT

Typical Container Characteristic	Typical Inspection Specifications
• Phenolic Plug	12.2 lbs. minimum weight of plug
• Submerged bubble pressure test for the inner container	15 PSIG minimum Prior to first use, leak tightness is verified by a submerged bubble pressure test at 15.0 PSIG, minimum. Submersion time is one minute minimum. Test is conducted using the silicone rubber container gasket as the only sealing agent between flange and cover.
• Verification of container measurements	Based on approved dimensions on licensed drawing
• Appearance integrity (Visual)	No visible holes or cracks, and no significant absence of paint.
• Boral Liner	Verification of dimensions based on licensing drawing Visual for physical integrity Visual at ends of liner for missing Boral material between stainless steel layers Review vendor test result for boron content to assure areal density Review certification for traceability of Boral to liner serial number
• Boron content in the foam	For BU-7s fabricated after 1993, assure that the minimum value of the * * *

Typical Container Characteristic	Typical Inspection Specifications
	boron in the foam exceeds the value used in the criticality safety analysis by at least 133% (this allows credit for no more than 75% of the boron verified as present in the foam). *

5.2.2 Failures are rejected and, where appropriate, reworked and retested.

5.3 MAINTENANCE PROGRAM

The following procedures represent the activities involved in the maintenance program for the BU-7 package.

5.3.1 The BU-7 outer drum and cover are inspected to assure:

- Good adherence of paint
- No visible holes or cracks in the metal surfaces
- No dents which affect drum integrity
- Closure rings and bolts are in good condition
- The four 1/4 inch holes in sides near top of drum are covered with weatherproof tape

- Phenolic foam insulation plug is in serviceable condition

- Container is appropriately marked

5.3.2 The BU-7 inner drum and lid are inspected to assure:

- Flange weld is intact

- Inner drum is visibly clean and painted

- Inner drum gasket sealing surface is clean, smooth and flat with no rust spots

- A new gasket is used, or the existing gasket is replaced with a new gasket if inspection shows any defects. The inner drum gasket must be changed if the gasket has been in service for more than 12 months at the time of packing

- Threads are in good condition

- There is no visible indication of holes

5.3.3 The BU-7 Boral liner is visually inspected to assure:

- The inner and outer layers of stainless steel do not have holes or punctures other than allowed by the drawing

- There is no evidence that the Boral sandwiched in between the stainless steel is missing

- Weld integrity

5.4 Criteria for Repair or Replacement of Container Components

When components parts of the BU-7 packaging do not meet the maintenance program inspection criteria, they are either reworked or replaced.

6.0 BU-7 TRANSPORT PACKAGE SPECIFICATIONS

Specifications for the BU-7 transport package are shown on General Electric Drawing 112D1592 in Appendix A.

GE Nuclear Energy

Nuclear Fuel & Components Manufacturing
General Electric Company
P O. Box 780, Wilmington, NC 28402
919 675-5000

December 14, 1993

Mr. Charles J. Haughney
Transportation Certification Branch
Division of Fuel Cycle & Material Safety
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Haughney:

References: (1) Consolidated Application for BU-7 Package, C.M. Vaughan to C.J. Haughney, 12/3/93, Submitted on 12/10/93
(2) Telephone Conversation of 12/9/93, C.M. Vaughan et al GE and C.J. Haughney et al US NRC

On December 10, 1993, we submitted to the NRC our consolidated application for the BU-7 package, dated December 3, 1993. GE is firmly convinced that our consolidated application contains sufficient information to demonstrate that the BU-7 package is acceptable for shipment under the proposed conditions and limits in accordance with the regulatory requirements of 10CFR71. This is also consistent with the NRC position prior to our application of December 6, 1991 that extended the enrichment range for the package to 5% U-235.

Although the consolidated application conforms to the NRC regulations, including providing the regulatory required information and test data in accordance with the regulatory defined testing process, we understand from our conversation on December 9 that the NRC is currently concerned about the appropriateness of moderation exclusion for the BU-7 package because of questions related to the robustness of its design.

As indicated in our transmittal letter dated December 3, 1993, GE had completed a comprehensive internal review of the compliance and safety basis for the BU-7 package before submitting the newly consolidated application to the NRC. This review generated an extensive volume of information that supports the safety of the container design and performance under testing. Since our telephone conference on December 9 indicated that this information would assist the NRC in its review of the consolidated application, summaries of this information are being provided to the NRC as described below.

Attachment A summarizes engineering design verification calculations (Buckling Analysis of Fuel Shipping Cannister, Dias to Baumgartner, December 3, 1993) that verify the structural adequacy of the internal 16-gallon drum. The results of our simplified handbook approach are comparable to the NRC's treatment using ASME Tables. Of principle importance, however, a finite element buckling analysis demonstrated a minimum safety factor of 15%. These results are confirmatory of the design and clearly predicts the acceptable test results and performance that has been observed.

Attachment B (summary of calculations and test results, Kaul to Baumgartner, November 30, 1993) provides a structural evaluation of the inner product pail used within the BU-7. GE's consolidated application includes test data and photographs which demonstrate that the loaded pails passed the 10 meter accidental drop requirement. The calculations in this work demonstrate that passing the test is consistent with the strength of the pail. Therefore, assuming widespread distribution of powder outside the pail in an accident is not consistent with the strength margin and demonstrated performance of the pail.

Attachment C (BU-7 Hydro Test Data, Baumgartner to Vaughan, November 29, 1993) summarizes the BU-7 Hydro Test Data performed on 424 BU-7 containers in 1982/83 as a part of our effort to register the BU-7 package for use in Japan. This is a sizable test population that demonstrates that the container is robust enough to withstand hydrostatic pressures equivalent to 50 feet submersion. The test reports are available for inspection if necessary. Additionally the 16-gallon drums were robust enough at this test pressure so that they were not buckled or otherwise deformed so as to impact their assembly into the specification container.

In addition, as mentioned in our telephone conversation, the BU-J container is nearly identical to the BU-7 and in fact uses a metric equivalent of the inner 16-gallon drum. In recently purchasing and leasing a group of 431 of these BU-J containers, we determined that they came from a parent population of 1,280 fabricated as a group. Each of the 1,280 16-gallon inner drums were double tested to the equivalent pressure of 50 feet (one test with internal pressure and one test with external pressure). To pass they all had to be leak tight. The test records are voluminous and therefore not included in this transmittal but are available for inspection. This clearly underscores the leak tightness of the inner container. In addition the 16-gallon drums were robust enough to withstand the pressure tests and were not buckled or deformed so as to fail to meet the specifications for the container.

There was also an indication during our December 9 teleconference that the interest in increased safety margin came from moving to enrichments in the 4-5% U-235 range. To the contrary, our criticality safety analysis indicates that for normal cases the K_{eff} is fairly flat as a function of enrichment (2 safe batches/container) and that in the accident case reactivity is higher at lower enrichments primarily because the masses of powder are larger.

Again let me say that GE strongly believes that moderation exclusion is an appropriate assumption in demonstrating the safety of the BU-7 package in accordance with NRC regulatory requirements. This is clearly supported by design and test data. We are prepared to work with the NRC in promptly resolving these matters, which are of vital importance to GE.

Sincerely,

GE NUCLEAR ENERGY

C. M. Vaughan

C. M. Vaughan, Manager
Regulatory and EHS

Attachments

/pl

cc: CMV-93-123

ATTACHMENT A

BUCKLING ANALYSIS

December 3, 1993

To: J. Baumgartner

cc: S. Ranganath
M.L. Herrera
R. Strine
D. Drendel
T. Dunlap
R.B. Elkins
M. Kaul

From: K.P. Dias K.P. Dias 12/3/93

Subject: Buckling Analysis of Fuel Shipping Cannister

Scope and Background

Buckling of a fuel shipping cannister under hydrostatic pressure was analyzed. Sketches of the cannister are shown in Figures 1 and 2. The cannister is sealed at the top and bottom essentially; a "thick" cover is bolted on the top and a thin walled bottom plate is attached at the bottom using a crimped joint. The specified material thickness is 0.0478 ± 0.005 inch for the cylinder and the bottom. However, the cover is 3/16" or 0.1875" thick (approximately 4 times the thickness of the thin-walled cylinder). The cannister also has two circumferential ribs for structural reinforcement as shown in Figure 2.

The specified hydrostatic design pressure is 21.7 psi, which is equivalent to 50 feet of water. Therefore, the cylinder must remain leak tight and maintain structural integrity up to 21.7 psi hydrostatic.

Simplified Handbook Prediction

A theoretical or handbook solution (Ref. Roark's *Formuas for Stress and Strain)* was initially used to calculate a buckling pressure of approximately 20.2 psi for minimal material thickness conditions using the following formula:

$$P_{cr} = 0.92\,E / [(\ell/r)\cdot(r/t)^{2.5}]$$
$$= 53{,}350\,(t)^{2.5}$$

where, E = elastic modulus = 29×10^6 psi
r = cannister radius = 7" (approx.)
t = material thickness = 0.0428 (min.)
ℓ = cannister length = 27" (approx.)

This critical pressure is associated with a cylindrical buckling mode and assumes the ends are constrained to remain circular. This is a reasonable assumption since the ends are constrained by circular plates on both top and bottom. However, the handbook formulation does not account for the circumferential ribs, which can add substantial circumferential and axial stiffness to the cannister.

Roark also adds that experimental critical pressures vary ± 20% about handbook theoretical values, indicating that eigenvalue predictions are adequate for estimating the actual buckling loads (imperfections and large deflection behavior inclusive).

<u>ANSYS Finite Element Model (FEM)</u>

To account for the added stiffness of the ribs, a detailed finite element analysis (FEA) of the fuel shipping cannister was performed. The model is shown in Figures 3 and 4. This model includes the following details which were not accounted for in the handbook solution:

- Circumferential ribs (using as measured dimensions)
- Non-rigid end constraints at the top and bottom
- Simulation of crimping and overlap weld (top) using local element thickness increase (= 3t, conservative)

A half-symmetry model was used (as shown) such that full mode shapes (wave modes) could be observed. Elastic quadrilateral shell elements (STIF63) with stress stiffening capability were used.

Figure 5 shows a cannister model without the rib features; all other geometry details are retained. Both models were analyzed; the latter was used to baseline the finite element model against the handbook formulation and provide a consistent basis for evaluating an increase in buckling load due to rib reinforcement.

Although a large deflection analysis is preferable, preliminary FEA using large deflection formulations yielded questionable results. Several problems with convergence were encountered. ANSYS convergence methods (Newton-Raphson techniques) can be problematic and do not always yield reliable solutions. Therefore, analyses (to date) have been restricted to linear eigenvalue buckling analyses.

<u>Finite Element Buckling Analysis</u>

Linear elastic eigenvalue buckling analysis is a "bifurcation" analysis and usually provides *upper bounds* of buckling loads or limit loads. The finite element eigenvalue formulation is consistent with "classical" buckling solution methods in that the results are dependent upon the *original* geometry and do not account for large pre-buckling deformations. Therefore, it is important to note that both handbook calculations and the subsequent linear FEA buckling evaluation represent upper bounds on buckling loads. However, the analytical results provided here, coupled with test data, should provide a sound basis for assessing the limit load capability of the fuel shipping cannisters.

The first buckling mode associated with the collapse of the cylinder wall is shown in Figures 6 and 7. Note that the cylinder ends not only remain circular but are also closely approximate a constrained end condition --- a condition implicit in the handbook formulation. The ribs take on an elliptical buckling shape characteristic of rings under external pressure as would be expected.

Finally, as a comparison, the first cylindrical buckling mode for the non-ribbed cannister is shown in Figure 8. Once again, the ends remain circular and approximate a constrained end condition.

All cases were run using the minimum thickness of 42.8 mil for the cylinder and a maximum thickness of 52.8 mil for the bottom base plate of the cylinder. Thickness variations in the base alone are not expected to have a significant impact on the buckling mode of the cylinder. No finite element cases were run for increased thicknesses of the cylinder. However, the analytical buckling forumula for a cylinder can be readily used to obtain a reasonable estimate of the effect of thickness on buckling load, since the effect of thickness on buckling loads should be approximately the same for both configurations (i.e. with and without ribs).

The following results were obtained:

	With Ribs	Without Ribs
@ 42.8 mil: (min)		
ANSYS	35.5 psi	23.0 psi
Handbook	n/a	20.2 psi
@ 47.8 mil: (nom.)		
ANSYS*	(46.9 psi)	(30.4 psi)
Handbook	n/a	26.7 psi
@ 52.8 mil: (max)		
ANSYS*	(59.9 psi)	(38.8 psi)
Handbook	n/a	34.1 psi

* Values obtained by scaling 35.5 psi, given theoretical relation of $P_{cr} \propto t^{2.5}$.

The minor discrepancy between the handbook value of 20.2 psi and the ANSYS value of 23.0 psi can be attributed to two things:

1. The element mesh size is not refined enough; larger mesh sizes generally overpredict critical buckling loads.
2. Thickness is increased (3t) locally over an 11/16" length at top and bottom, thus reducing the effective length (for FEM). A minor increase in the critical pressure can be expected.

Nonetheless, the degree of accuracy is sufficient to validate the use of this model for estimating eigenvalue buckling loads and evaluating the added effect of rib reinforcement.

As such, using the ANSYS finite element results to account for the effect of rib reinforcement, the effective increase in buckling capability can be calculated as follows,

$$35.5 \text{ psi} / 23 \text{ psi} = 1.54$$

To ensure conservatism, this factor can be applied to the lower handbook value of 20.2 psi to calculate an equivalent theoretical load for a rib reinforced cannister,

$$P_{cr} = 1.54 \times 20.2 \text{ psi} = 31.1 \text{ psi}$$

If one were to follow the recommendation of Roark, the 35.5 psi critical pressure could be further reduced by 20%,

$$P_{cr,\ lower\ bound} = 0.8 \times 31.1 \text{ psi} = \underline{24.9 \text{ psi}}$$

Based upon the combination of test results and the above analytical predictions, it would seem that the lower bound critical pressure of 24.9 psi is reasonable. Test results showed no permanent buckling deformation at 21.7 psi (50 feet of water); thus, it is expected that the limit load should be higher than 21.7 psi. The ***minimum*** factor of safety is then,

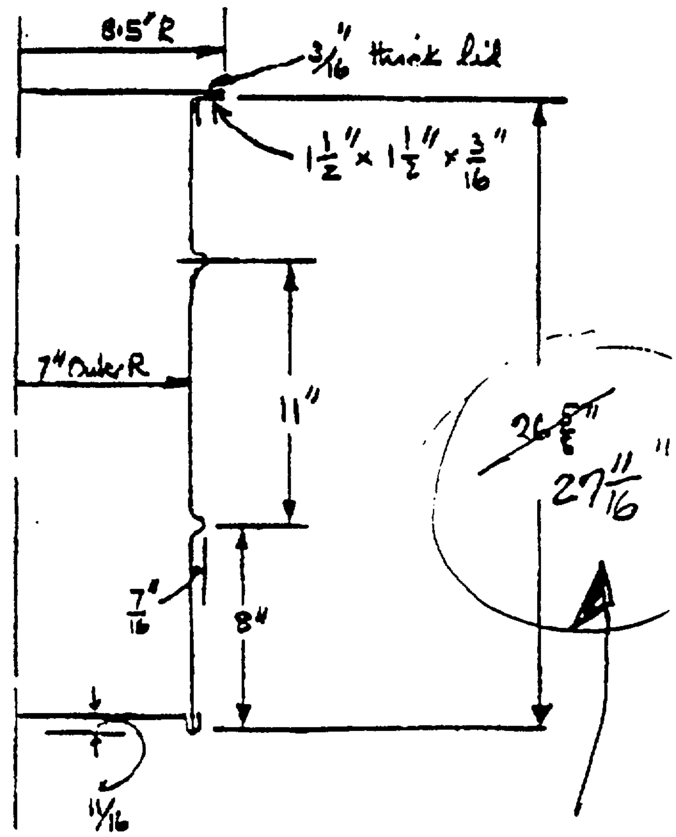
$$\text{F.O.S.} = 24.9/21.7 = 1.15$$

Finally, it should be noted that several additional conservatisms are inherent in the above analysis. The fuel shipping cannister above has been analyzed as an *empty and evacuated* cylinder subjected to a direct external hydrostatic pressure. In actuality, the inner cannister or drum is:

- filled with fuel powder
- suspended in a phenolic foam insulation which is formed in place between the outer drum and inner drum.

Both of these effects will *substantially* increase the rigidity and resistance to deformation of the inner cannister under what is truly hydrostatic loading of the outer drum. The phenolic foam will provide added constraint and structural support such that large deflections should be significantly restricted in comparison to the model used in the analysis above. This will increase the load carrying capability and provide additional safety margin.

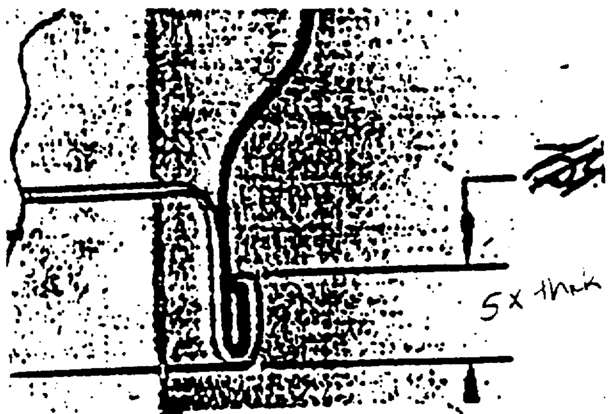
Inner Drum of 18 gage
Steel (0.0478" thick)
Type: A109 Carbon Steel
Bottom sole and drum
wall of same thickness



1 Change Only

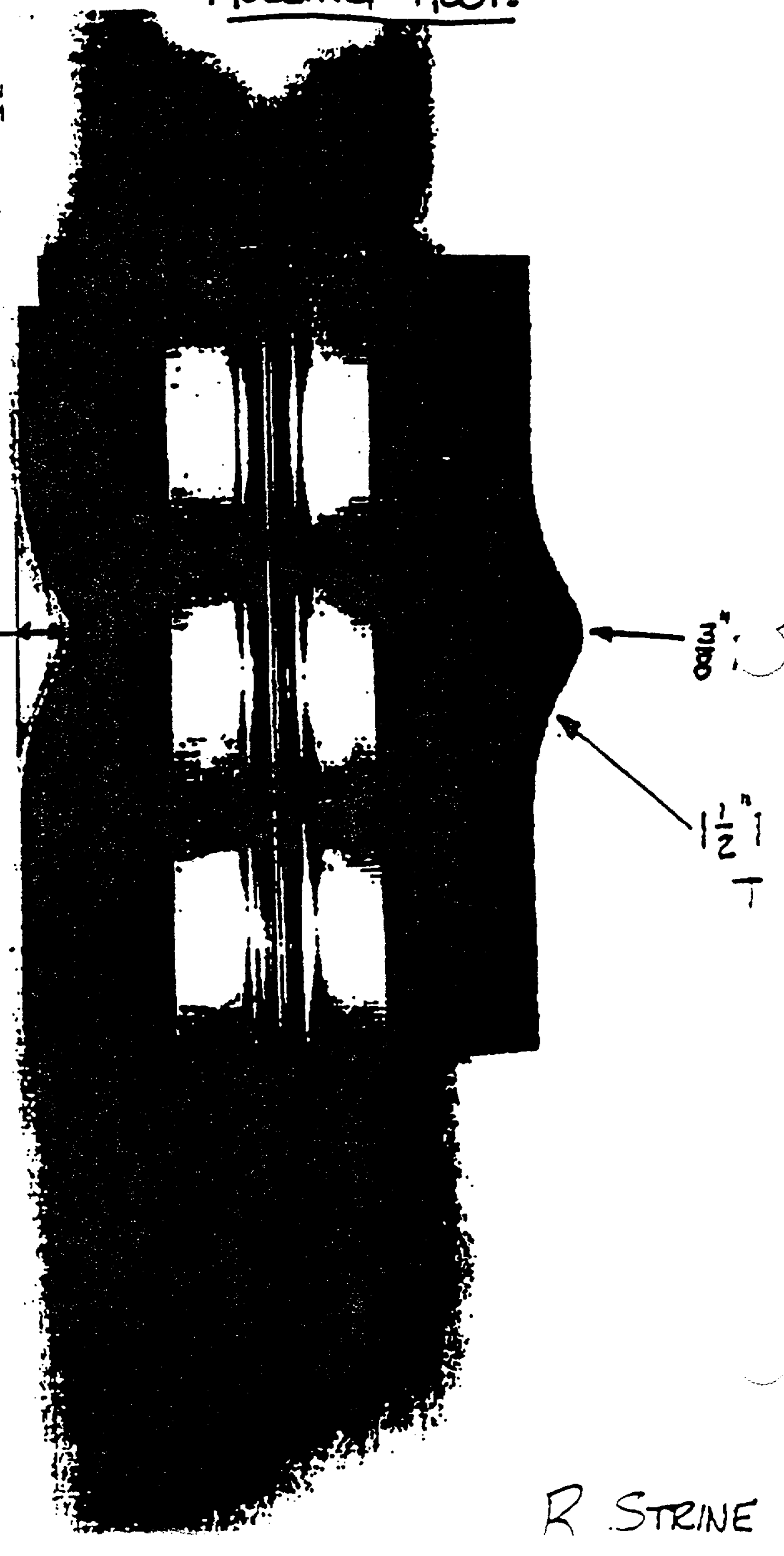
NEED 27" [cut off]
HEIGHT INSID[cut off]
THE INNER
CONTAINER [cut off]
THE LID O[cut off]

FIGURE



Bottom To Shell
Mechanical Connection.
5 Layers Of Material
Thick.

Rolling Hoop.



1. Vendor And
Actual Measurements
Made By Myself
Agree On Hoop
Dimensions
2. Bottom Is
18 Gauge Also

Figure 2

R Strine

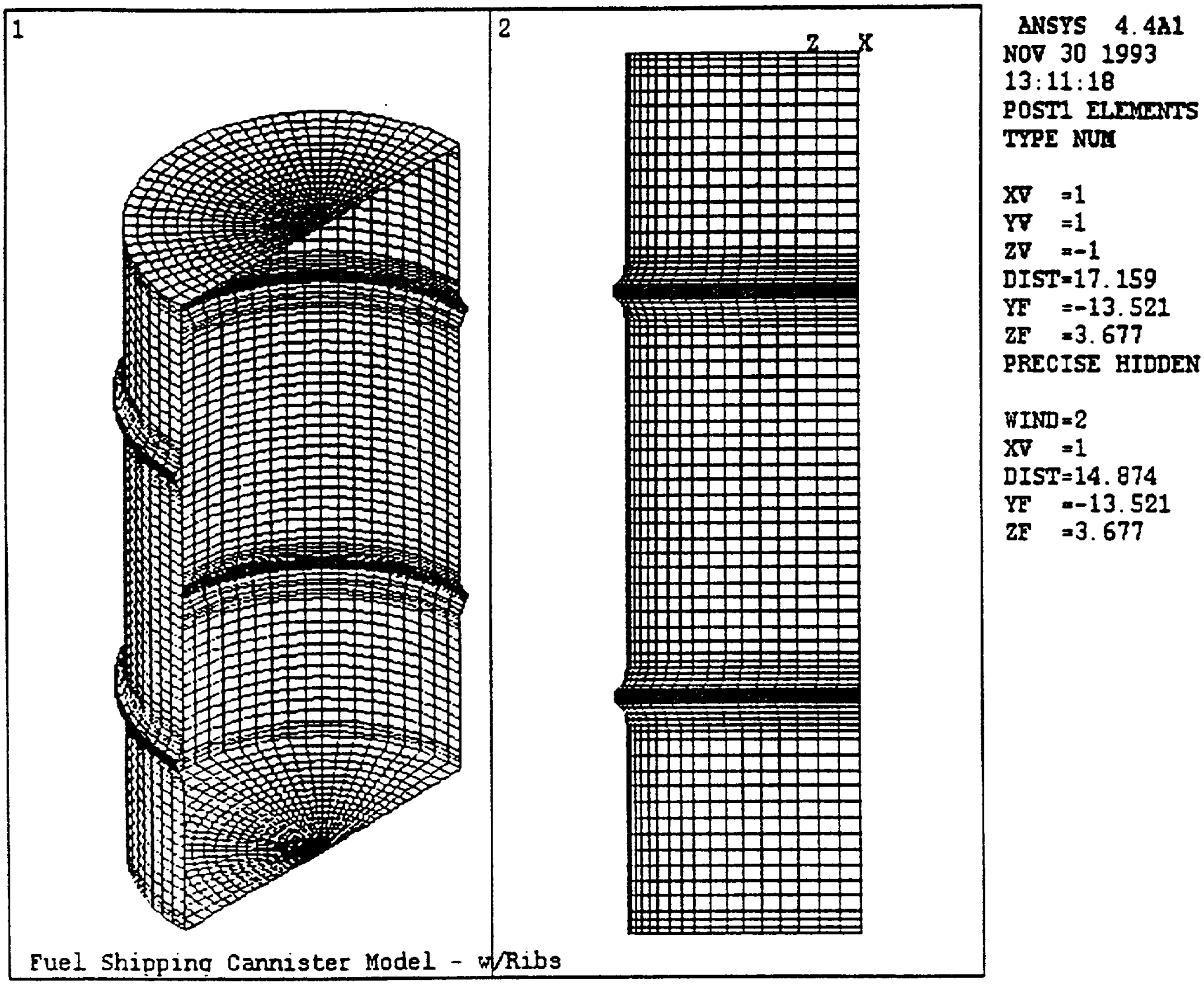


Figure 3. ANSYS finite element model of fuel shipping cannister shown with ribs.

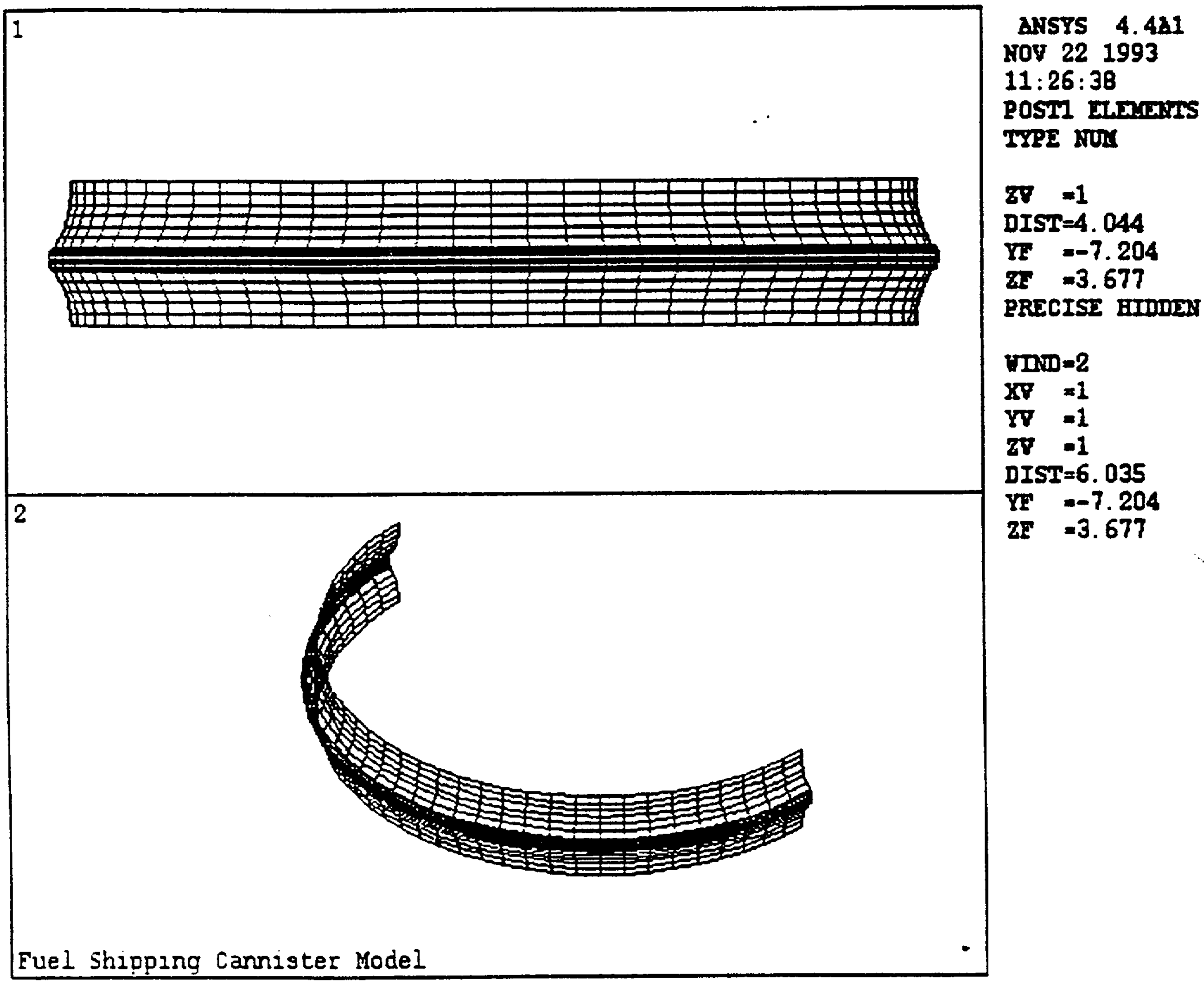


Figure 4. Detail of rib section of finite element model.

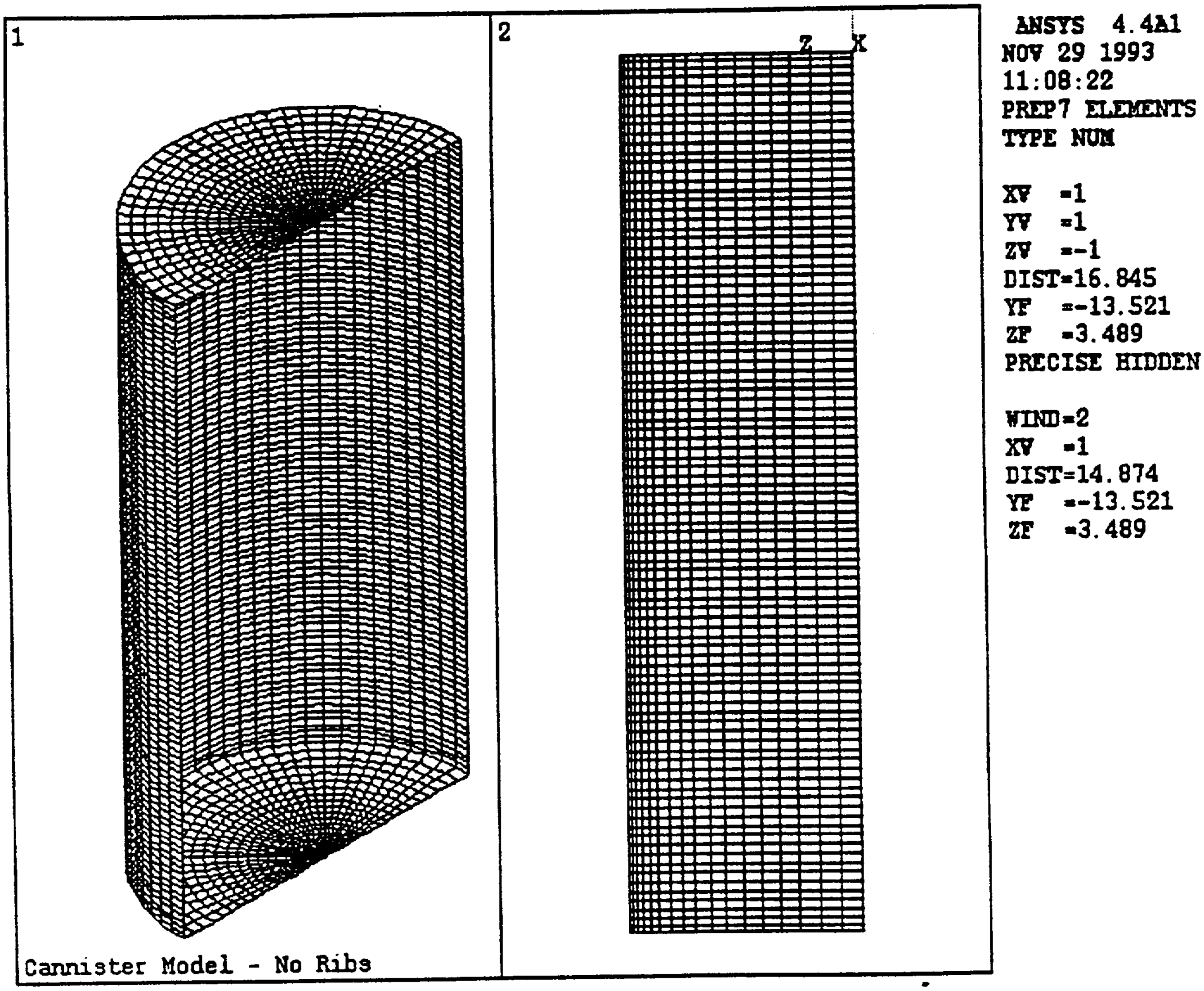


Figure 5. Finite element model of a simple cannister without rib reinforcement.

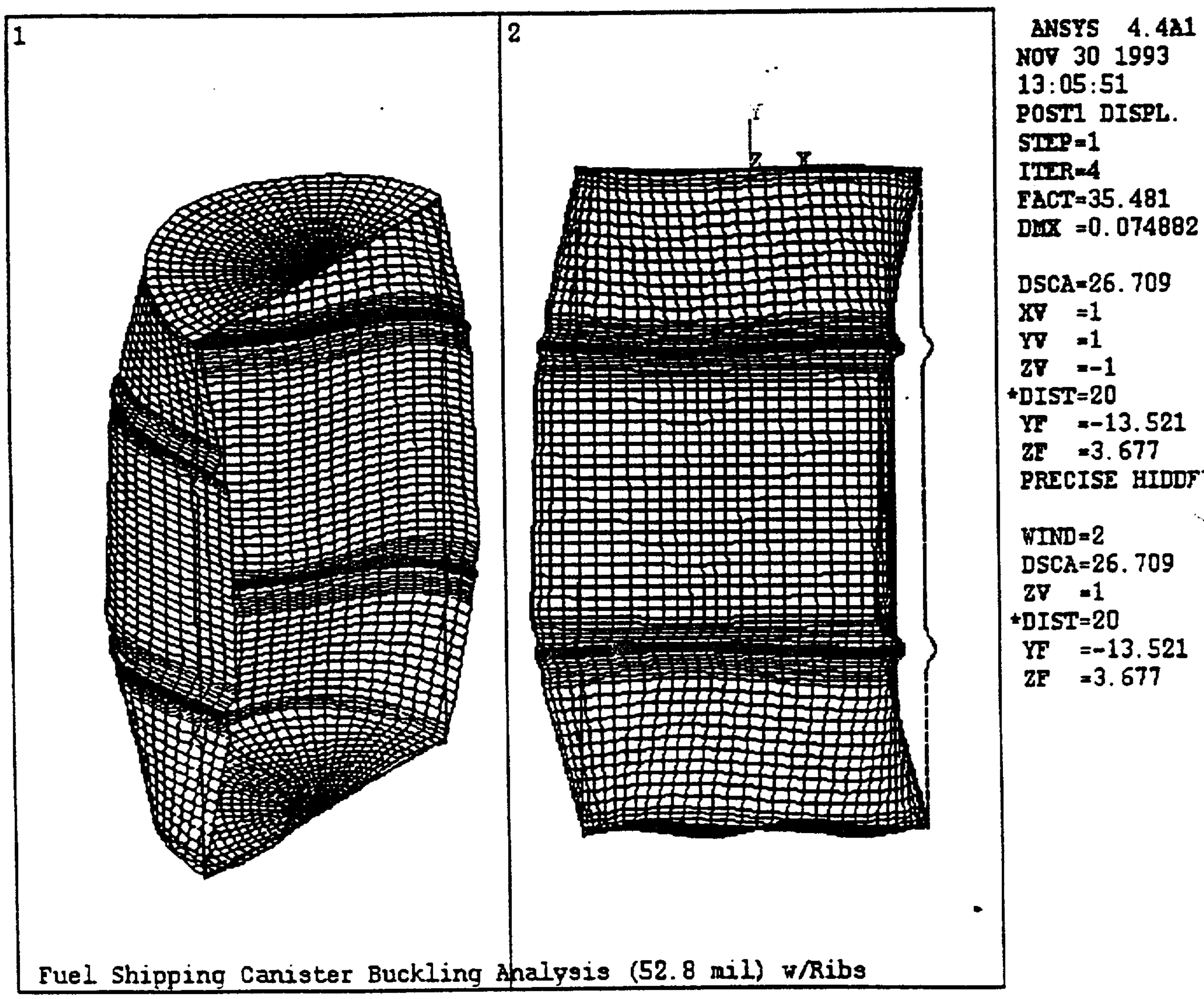


Figure 6. First eigenvalue buckling mode for ribbed shipping cannister (FEA result).

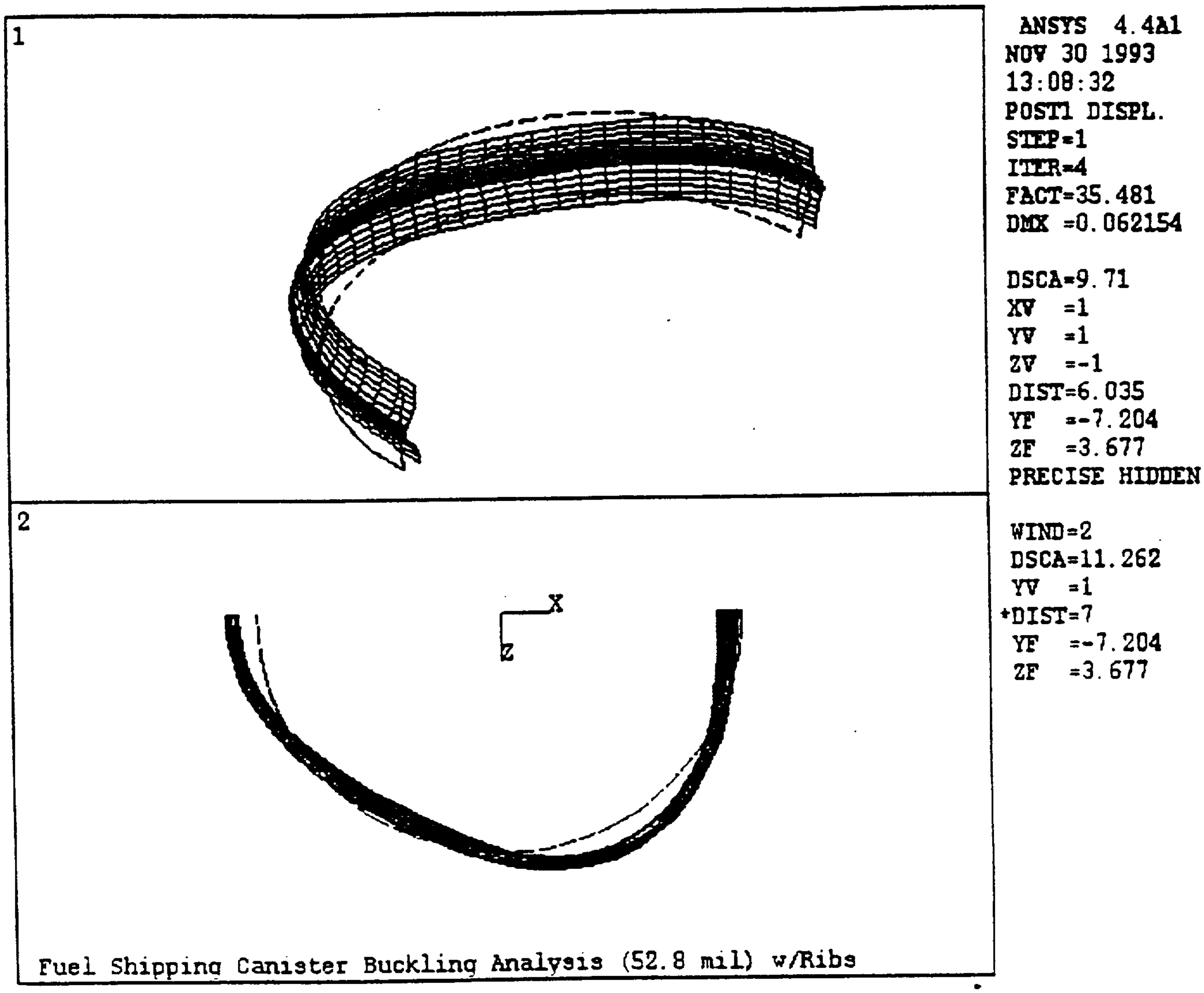


Figure 7. Detail of rib deformation (elliptical) for first buckling mode (FEA result).

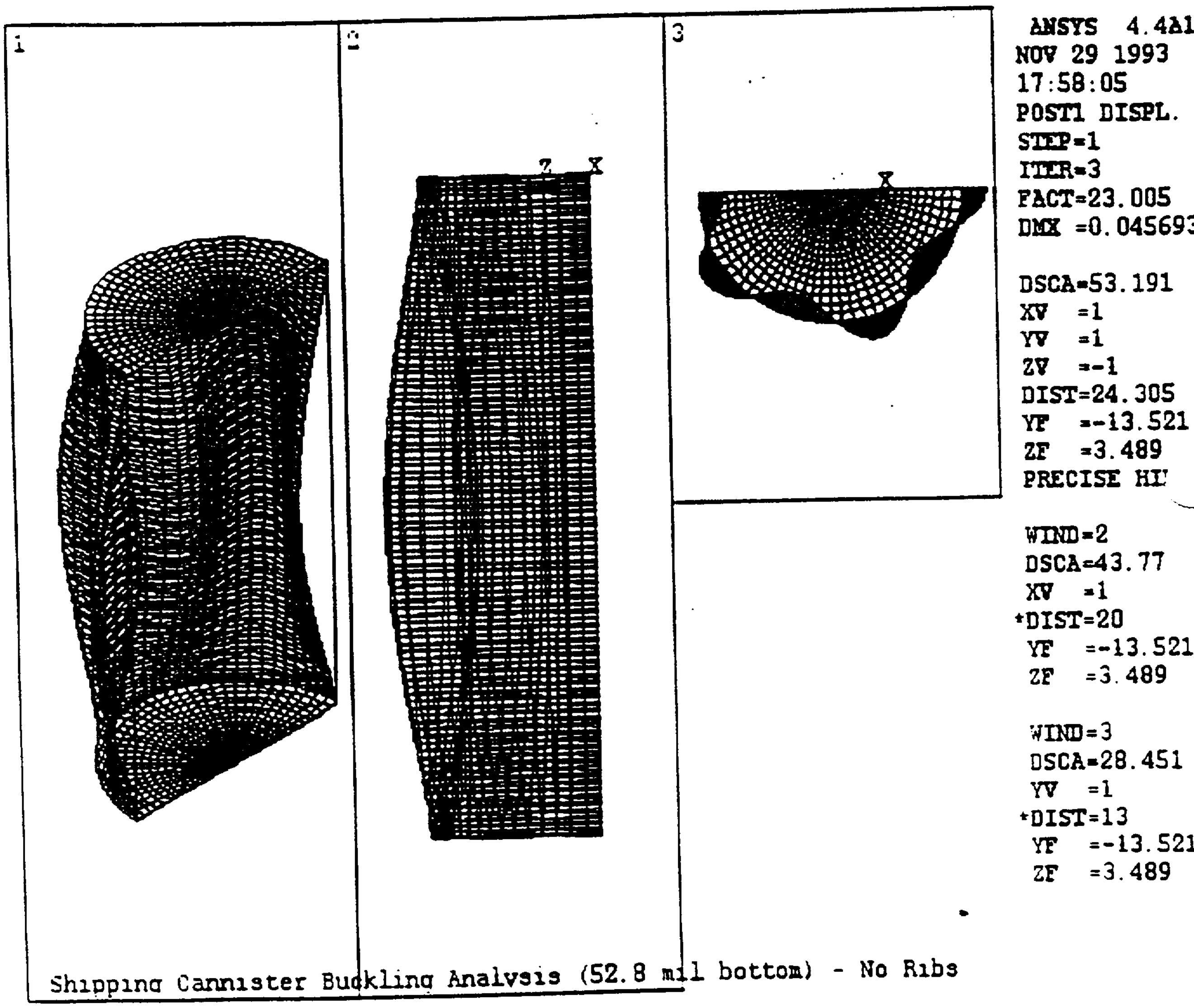


Figure 8. First buckling mode of cylindrical cannister without rib reinforcement (FEA result).

ATTACHMENT B

STRUCTURAL EVALUATION OF PRODUCT PAILS

November 30, 1993

TO:	John Baumgartner	M/C J09	X5821
FROM:	Maharaj Kaul	M/C 747	X3221

Enclosed are:

1. Summary of calculations and results demonstrating BU-7 integrity under a 10.5m accidental drop (DOT requirement is only 9 m, but JNF has been using 10.5 m). This package was prepared for the August 16, 1993 meeting with NRC.

2. Sample calculation on the pail.

3. Buckling calculation for the Boral Liner.

Please let me know if there is anything else that is required.

Mkaul

BU-7 CONTAINER ANALYSIS UNDER A 10.5 METER HYPOTHETICAL DROP

CONTAINER DESCRIPTION

GEOMETRY

The BU-7 container is made of two drums -- the 55-gallon outer drum and the 16 gallon inner drum. The outer drum conforms to the size specification of ANSI MH2.2 - 1979

Item	Metal Guage	Thickness (in.)
Outer Drum Body	18	0.0478
Outer Drum Lid	16	0.0598
Outer Drum Closure Ring	12	0.1046
Inner Drum	16	0.0598

MATERIAL PROPERTIES

The material properties of the outer and the inner drum are tabulated below.

Material Property	Outer Drum	Inner Drum
ASTM Steel Designation	A569	A570 Grade B
Yield Stress	22.0 ksi	----
Tensile Strength	----	49.0 ksi
Shear Strength	----	24.5 ksi

The total weight of the container exclusive of the uranium fuel and the pails ranges between 150 lbs. and 165 lbs.The uranium fuel weighs 154 lbs. and the containing pails have a weight of 16 lbs.

The inner drum lid is held by 12 7 /16 in. diameter UNC bolts of SAE Grade 1 with 14 threads per inch. The tensile area of each bolt is 0.1063 sq. in. and the tensile strength of the bolts is 60 ksi.

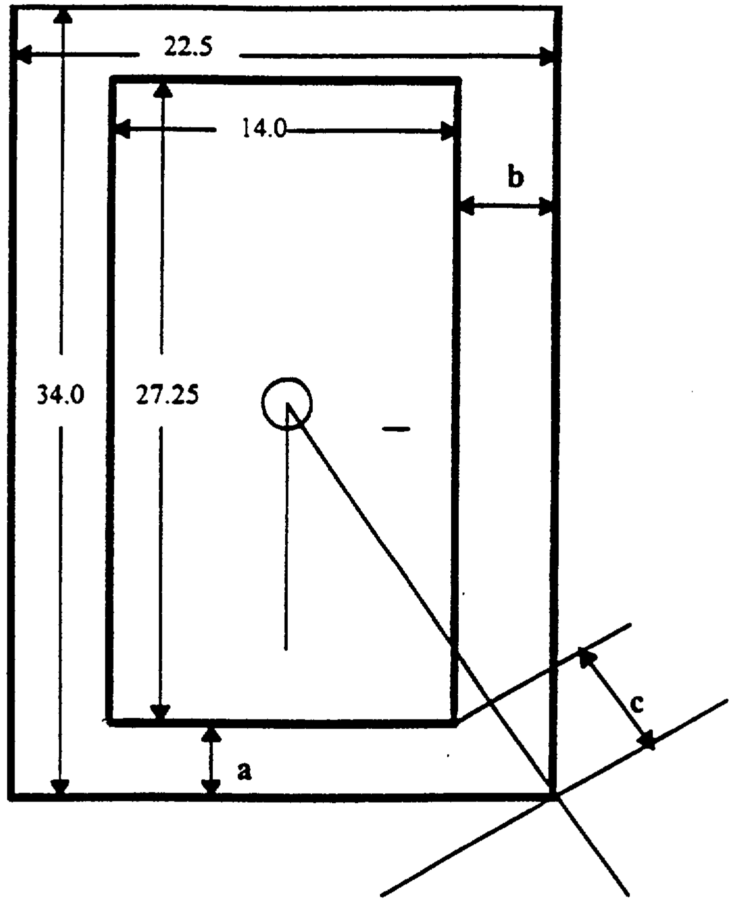


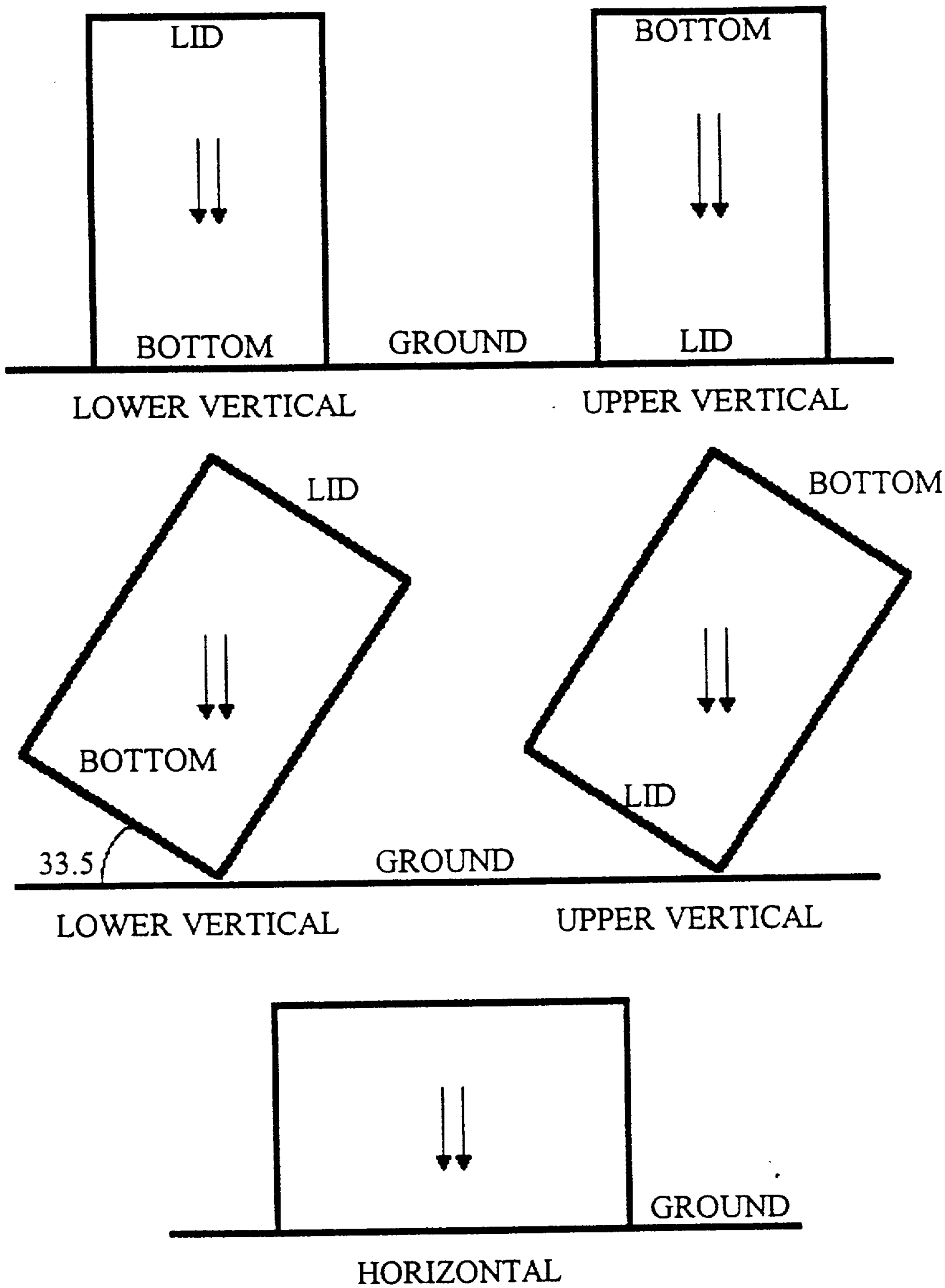
Figure 3.1 Relative Position of Drums in the Container

$a = (34.00 - 27.25) / 2 = 3.375 \; in$

$b = (22.50 - 14.00) / 2 = 4.250 \; in$

$\theta = \tan^{-1}(22.5/34.0) \quad = 33.5°$

$c = a.\cos\theta \; + \; b.\sin\theta \; = \; 5.160 \; in$



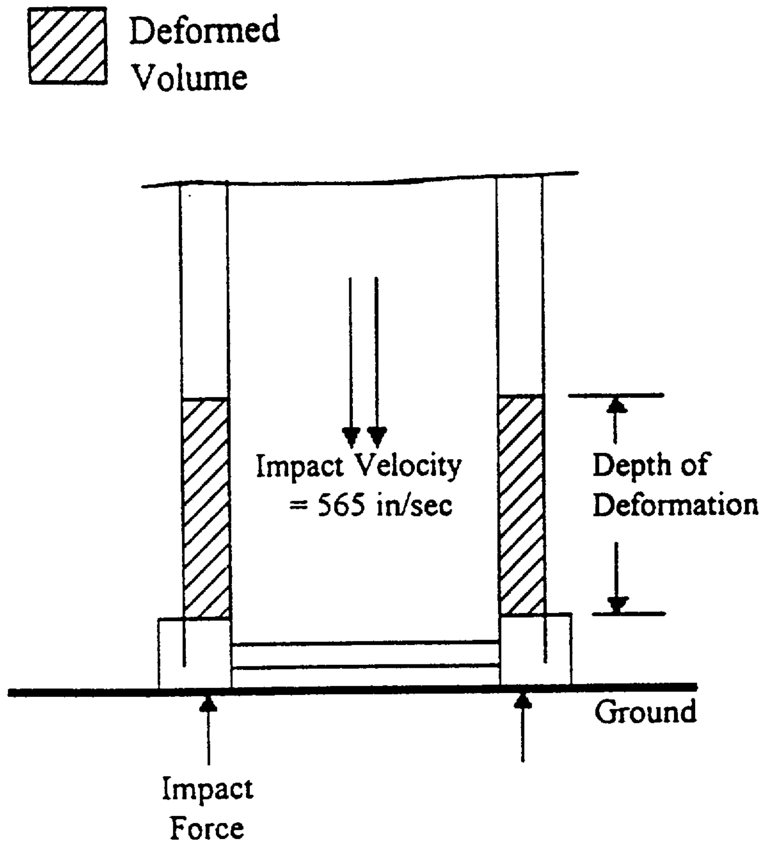
CONFIGURATIONS ANALYZED

ASSUMPTIONS

1. On impact, the kinetic energy of the container is completely absorbed by plastic deformation of the outer drum. No credit is taken for energy loss due to other reasons, such as

   a) Ground Deformation
   b) Ground Radiation damping
   c) Viscous Damping of the Container

2. For Inner Drum, Boral Liner and Pail calculations, the cushioning effect of the packaging foam is completely ignored. This assumption is extremely conservative. Simplified calculations show that if the period of oscillation associated with the inner drum and the foam is as small as 10 times the time it takes the container material to absorb all its kinetic energy, the cushioning foam will attenuate the outer drum peak acceleration to 43 percent of its value.

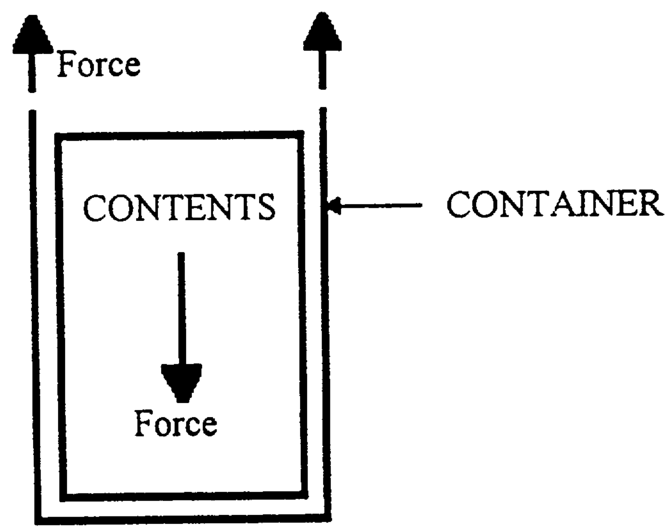


DEFORMATION MODEL

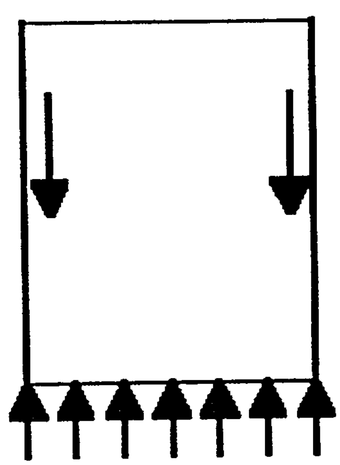
OUTER DRUM CALCULATIONS

	GEOMETRY AT IMPACT AFTER 10.5 M. DROP				
	Upper Vertical	Lower Vertical	Upper Corner	Lower Corner	Horizontal
Computed Depth of Deformation	1.840 in	1.840 in	1.704 in	2.370 in	1.410 in
Clearance Limited Deformation	3.375 in	3.375 in	5.160 in	5.160 in	4.250 in
Computed Maximum Acceleration	225 g	225 g	272 g	243 g	399 g

POTENTIAL FAILURE MODES OF INNER DRUM, PAIL AND LINER



SHEARING FORCE ON CONTAINER LID



BUCKLING FORCE ON LINER

LIMITS ON ACCELERATION

	MAXIMUM PERMISSIBLE ACCELERATION				
Limiting Parameter	Upper Vertical	Lower Vertical	Upper Corner	Lower Corner	Horizontal
Inner Drum Strength	450 g	384 g	540 g	460 g	966 g
Pail Strength - 24 Gauge	268 g	268 g	321 g	321 g	
Pail Strength - 20 Gauge/16 Lid	671 g	403 g	805 g	483 g	
Liner Buckling Strength	484 g	484 g	580 g	580 g	
Computed Maximum	225 g	225 g	272 g	243 g	399 g

URANIUM PAIL INTEGRITY

Assume that the maximum acceleration a that the pails and the contents experience when the 55 gallon container comes to rest is same as that of the container. This is a very conservative assumption as it completely ignores the cushioning effect of foam between the outer and the inner drums. Depending on geometry of container on impact, the force necessary to bring the pail-contents to rest will be applied on it through the pail lid or bottom. The same magnitude of force will also act on the pail and tend to shear the lid (or the bottom) off the pail.

The force F required to shear off the bottom is

$$F = 2\pi r t \tau$$

in which

r = radius of pail bottom = 5.625"
t = thickness of pail bottom = 0.0239" (24 gage)
τ = shear strength of pail bottom = 24.5 ksi (A570 Steel)

Therefore F =20,695 lbs.

The maximum acceleration of the fuel that the pail can withstand is

$$a = F / W$$

where W is the weight of fuel inside the pail and is 77 lbs. Therefore

$$a = 20695 / 77 = 268 \text{ g}$$

The maximum acceleration experienced by the container is 225 g.

The effect of two pails -- one on top of another -- does not seem to create a worst failure mode in spite of the increased weights.

For the following calculations, the outer drum maximum acceleration is assumed to be 225g which is computed by analyzing a 10.5 meter accidental drop of the container.

BUCKLING OF BORAL LINER

Roark (*Formulas for Stress and Strain*, 5th Edition, 1975) has formulas for buckling stresses for different structural shapes in its section on Elastic Stability. On page 555, the formula for a thin-walled circular tube under longitudinal compression is given as follows:

$$\sigma' = \frac{E}{\sqrt{3}\sqrt{1-\nu^2}} \cdot \frac{t}{r} \qquad (1)$$

in which the meaning of various parameters and their values for 1100 aluminum are:

σ' =	Buckling Stress	
E =	Young's Modulus	= 9,800,000 psi
ν =	Poisson's Ratio	= 0.33
t =	Cylinder Thickness	
r =	Cylinder Radius	

Tests values are 40 to 60 percent of the above theoretical value. Assuming a 40 percent of the theoretical value for the buckling stress, therefore,

$$\sigma' = 2.4 \times 10^6 \frac{t}{r} \text{ psi} \qquad (2)$$

To calculate the maximum stress experienced by the aluminum sleeve inside the inner drum, assume that the sleeve comes to rest (after the container impacts the ground) with the same maximum acceleration experienced by the outer drum. Let this acceleration be a (g's). In the present case a =225 g's. The weight W of the sleeve and any attachments to it that have to be brought to rest is 28 lbs. The stress σ caused by this force on the sleeve cross-section, which is an annulus of radius r and thickness t is given by

$$\sigma = Wa/(2\pi r t) = 1003/(rt) \text{ psi} \qquad (3)$$

For buckling not to occur,

$$\sigma \leq \sigma'$$

implying thereby that

$$t \geq \sqrt{1003/2.4 \times 10^6} = 0.02''$$

This analysis considers an extreme situation. The actual Boral liner is a composite section with innermost and outermost linings of steel of thickness 0.03" sandwiching two aluminum liners of 0.01" and a 0.08" thick column of Boron. This section is much stronger than the one required to resist buckling from an acceleration of 225 g.

URANIUM PAIL INTEGRITY

Assume that the maximum acceleration a that the pails and the contents experience when the 55 gallon container comes to rest is same as that of the container. This is a very conservative assumption as it completely ignores the cushioning effect of foam between the outer and the inner drums. Depending on geometry of container on impact, the force necessary to bring the pail-contents to rest will be applied on it through the pail lid or bottom. The same magnitude of force will also act on the pail and tend to shear the lid (or the bottom) off the pail.

The force F required to shear off the bottom is

$$F = 2\pi r t \tau$$

in which

r = radius of pail bottom = 5.625"
t = thickness of pail bottom = 0.0239" (24 gage)
τ = shear strength of pail bottom = 24.5 ksi (A570 Steel)

Therefore F =20,695 lbs.

The maximum acceleration of the fuel that the pail can withstand is

$$a = F / W$$

where W is the weight of fuel inside the pail and is 77 lbs. Therefore

a = 20695 / 77 = 268 g

The maximum acceleration experienced by the container is 225 g.

The effect of two pails -- one on top of another -- does not seem to create a worst failure mode in spite of the increased weights.

For the following calculations, the outer drum maximum acceleration is assumed to be 225g which is computed by analyzing a 10.5 meter accidental drop of the container.

BUCKLING OF BORAL LINER

Roark (*Formulas for Stress and Strain*, 5th Edition, 1975) has formulas for buckling stresses for different structural shapes in its section on Elastic Stability. On page 555, the formula for a thin-walled circular tube under longitudinal compression is given as follows:

$$\sigma' = \frac{E}{\sqrt{3}\sqrt{1-\gamma^2}} \cdot \frac{t}{r} \qquad (1)$$

in which the meaning of various parameters and their values for 1100 aluminum are:

σ' =	Buckling Stress	
E =	Young's Modulus	= 9,800,000 psi
ν =	Poisson's Ratio	= 0.33
t =	Cylinder Thickness	
r =	Cylinder Radius	

Tests values are 40 to 60 percent of the above theoretical value. Assuming a 40 percent of the theoretical value for the buckling stress, therefore,

$$\sigma' = 2.4 \times 10^6 \frac{t}{r} \text{ psi} \qquad (2)$$

To calculate the maximum stress experienced by the aluminum sleeve inside the inner drum, assume that the sleeve comes to rest (after the container impacts the ground) with the same maximum acceleration experienced by the outer drum. Let this acceleration be a (g's). In the present case a =225 g's. The weight W of the sleeve and any attachments to it that have to be brought to rest is 28 lbs. The stress σ caused by this force on the sleeve cross-section, which is an annulus of radius r and thickness t is given by

$$\sigma = Wa/(2\pi r t) = 1003/(rt) \text{ psi} \qquad (3)$$

For buckling not to occur,

$$\sigma \leq \sigma'$$

implying thereby that

$$t \geq \sqrt{1003/2.4\times 10^6} = 0.02''$$

This analysis considers an extreme situation. The actual Boral liner is a composite section with innermost and outermost linings of steel of thickness 0.03" sandwiching two aluminum liners of 0.01" and a 0.08" thick column of Boron. This section is much stronger than the one required to resist buckling from an acceleration of 225 g.

ATTACHMENT C

BU-7 HYDRO TEST DATA

GE NUCLEAR ENERGY
Nuclear Fuel
8* 675-5821 M/C J09

November 29, 1993

cc: RF Calcaterra
RE Strine

To: CM Vaughan

Subject: BU-7 Hydro Test Data

During 1982 and 1983, GE took delivery of new BU-7 containers from Precision Metal Products, Inc. Our records show that 424 of these BU-7 containers have documentation showing that hydrostatic testing of the inner containers was performed at the equivalent of a 50 foot submersion in water for a period of 8 hours, with no water in-leakage.

The attached table summarizes the hydro tested BU-7 containers by serial number. The containers are grouped by one of two GE Purchase Orders, 334-G2864 or 334-AM286, and by the shipping order number from the vendor. Documentation consists of a Certificate of Compliance for each of the shipping orders, stating that the integrity of the weld joints and gasket surfaces have been 100% tested pressure checked (to 21.4 psig minimum). In addition to the Certificates of Compliance, the majority of the containers have individual GE receiving inspection Quality Control Inspection Instruction (QCII) sheets indicating that the hydro test of the inner container was performed. Copies of each of the Certificates of Compliance and QCII sheets are available if needed.

John Baumgartner
Manager, Wilmington Engineering

BU-7 INNER CONTAINER HYDRO TESTS

PO NO.		334-G2864		334-G2864		334-G2864		334-G2864	334-G2864
SHIP ORDER NO.		27547		27551		27552		27592	27620
DATE SHIPPED		7/1/82		7/2/82		7/6/82		7/16/82	7/22/82
NO. TESTED		4		16		30		40	18
		Serial No.		Serial No.		Serial No.		Serial No.		Serial No.
1	K-	4194	K-	4135	K-	4134	K-	4485	K-	4585
2		4207		4238		4137		4490		4597
3		4199		4252		4166		4500		4654
4		4197		4259		4169		4504		4670
5				4276		4178		4508		4675
6				4314		4180		4512		4680
7				4320		4210		4515		4684
8				4325		4212		4518		4688
9				4326		4214		4520		4698
10				4332		4217		4526		4702
11				4362		4229		4532		4706
12				4382		4270		4536		4712
13				4390		4286		4540		4825
14				4394		4290		4546		4843
15				4397		4296		4551		4852
16				4405		4300		4556		4859
17						4306		4561		4867
18						4309		4567		4886
19						4339		4570		
20						4345		4576		
21						4351		4580		
22						4355		4662		
23						4358		4666		
24						4402		4718		
25						4425		4724		
26						4458		4726		
27						4467		4730		
28						4472		4736		
29						4476		4741		
30						4480		4752		
31								4755		
32								4772		
33								4776		
34								4779		
35								4802		
36								4806		
37								4810		
38								4816		
39								4820		
40								4496		

	334-G2864		334-G2864		334-G2864		334-G2864		334-G2864		334-G2864
	27644		27676		27677		27690		27705		27753
	7/23/82		7/28/82		7/29/82		8/4/82		8/6/82		8/13/82
	20		8		3		8		16		3
	Serial No.		Serial No.		Serial No.		Serial No.		Serial No.		Serial No.
K-	4694	K-	6110	K-	6168	K-	6194	K-	6188	K-	6294
	4831		6117		6176		6199		6226		6302
	4837		6125		6183		6202		6234		6526
	4838		6126				6210		6242		
	4839		6134				6217		6317		
	4840		6142				6225		6325		
	4841		6150				6250		6326		
	4842		6160				6260		6334		
	6026								6342		
	6034								6350		
	6042								6360		
	6050								6368		
	6060								6376		
	6068								6383		
	6076								6388		
	6083								6394		
	6088										
	6094										
	6099										
	6102										

	334-G2864		334-G2864		334-G2864		334-G2864		334-G2864		334-G2864
	27761		27803		27842		27868		27902		28128
	8/17/82		8/27/82		9/3/82		9/13/82		9/23/82		11/8/82
	12		4		9		2		27		4
									***		***
	Serial No.		Serial No.		Serial No.		Serial No.		Serial No.		Serial No.
K-	6268	K-	6476	K-	6276	K-	6450	K-	6399	K-	6676
	6299		6542		6283		6458		6402		6694
	6310		6550		6288				6410		6726
	6483		6560		6460				6417		6742
	6488				6568				6425		
	6494				6576				6426		
	6499				6583				6434		
	6502				6588				6442		
	6510				6594				6599		
	6517								6602		
	6525								6610		
	6534								6617		
									6625		
									6626		
									6634		
									6642		
									6650		
									6660		
									6668		
									6683		
									6688		
									6699		
									6702		
									6710		
									6717		
									6725		
									6734		

*** Separate Test Sheets are available

										TOTAL
	334-AM286		334-AM286		334-AM286		334-AM286		334-AM286	# OF BU7
	28429		28415		28497		28516		28542	HYDRO-
	2/15/83		2/16/83		3/11/83		3/18/83		3/28/83	TESTED
	88		24		48		20		20	424
	***		***		***		***		***	
	Serial No.		Serial No.		Serial No.		Serial No.		Serial No.	
K-	6986	K-	7074	K-	7098	K-	7146	K-	7166	
	to		to		to		to		to	
	7073		7097		7145		7165		7185	
All containers purchased per PO # 334-AM286 were subsequently renumbered										
	New Nos.		New Nos.		New Nos.		New Nos.		New Nos.	
K-	7428	K-	7516	K-	7540	K-	7588	K-	7608	
	to		to		to		to		to	
	7515		7539		7587		7607		7627	

*** Separate Test Sheets are available

GE Nuclear Energy

Nuclear Fuel & Components Manufacturing
General Electric Company
P.O. Box 780, Wilmington, NC 28402
919 675-5000

December 22, 1993

Mr. Charles J. Haughney, Chief
Transportation Certification Branch
Division of Fuel Cycle & Material Safety
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Haughney:

References:
(1) Consolidated Application for BU-7 Package, C.M. Vaughan to C.J. Haughney, 12/3/93, Submitted on 12/10/93
(2) Telephone Conversation of 12/9/93, C.M. Vaughan et al GE and C.J. Haughney et al US NRC
(3) Letter, C.M. Vaughan to C.J. Haughney, 12/14/93

This letter is to correct and clarify a statement in my letter to you dated 12/14/93. The last paragraph on page 2 states that each of the 1,280 16-gallon inner drums were pressure tested internally and externally at an equivalent pressure of 50 feet. There were in fact two different tests conducted on these inner drums; one with water pressure and one with air pressure.

The water pressure test was conducted with the force applied on the exterior of the inner drums to demonstrate the structural and sealing capability of the unit. The air pressure test was conducted with the force applied on the inside of the drum to demonstrate the sealing capability of the lid.

The exterior pressure test consisted of applying water pressure of 147kPa for 8 hours. This is equivalent to a 15 meter water submersion test.

The interior pressure test consisted of applying air pressure of 74kPa to the inside of the drum and verifying that no bubbles appear in a soap bubble snoop test. This pressure is equivalent to 7.6 meters of water pressure.

In both tests, all the BU-J inner drums passed and were shown to meet design specifications in that they were both structurally sound and leak tight.

I apologize for any inconvenience this may have caused in your review.

Sincerely,

GE NUCLEAR ENERGY

C.M. Vaughan

C. M. Vaughan, Manager
Regulatory and EHS

cc: CMV-93-125

IDENTIFICATION MARK
J/79/AF-85(Rev.1)

COMPETENT AUTHORITY
OF
JAPAN

CERTIFICATE OF APPROVAL OF PACKAGE DESIGN FOR THE TRANSPORT OF RADIOACTIVE MATERIALS

ISSUED BY

SCIENCE AND TECHNOLOGY AGENCY
2-2-1, KASUMIGASEKI, CHIYODA-KU
TOKYO, JAPAN

SCIENCE AND TECHNOLOGY AGENCY
PRIME MINISTER'S OFFICE

2-2-1 Kasumigaseki, Chiyoda-ku, Tokyo 100, JAPAN
Telephone: Tokyo (03) 3581-5271
Telex: 02226720 STASGDJ

科学技術庁
〒100 東京都千代田区霞が関2－2－1

CERTIFICATE OF APPROVAL OF PACKAGE DESIGN FOR THE TRANSPORT OF RADIOACTIVE MATERIALS

This is to certify, in response to the application by JAPAN NUCLEAR FUEL Co., Ltd. on November 21, 1994, that the design of package described herein satisfies the design requirements of type A fissile package specified in Regulations for the Safe Transport of Radioactive Material (International Atomic Energy Agency, Safety Series No.6, 1985 Edition).

COMPETENT AUTHORITY
IDENTIFICATION MARK : J/79/AF-85(Rev.1)

5 Sep. 1995
Date

for Katsuyo Watanabe
Masayasu Miyabayashi
Director General
Nuclear Safety Bureau
Science and Technology Agency
Competent Authority of Japan
for Transport Package Design
of Radioactive Materials

1. NAME OF PACKAGE : BU-J (Type A, Fissile)

2. SPECIFICATION OF CONTENT

(1) Description of Contents

(i) Material of Nuclear Fuel	:	Uranium Dioxide (Powder and Pellet)
(ii) Enrichment	:	5.0% or less
(iii) Isotopic Content	:	$^{232}U \leqq 0.002\ \mu g/g\ ^{235}U$
		$^{234}U \leqq 10,000\ \mu g/g\ ^{235}U$
		$^{236}U \leqq 5,000\ \mu g/g\ ^{235}U$
		$^{99}Tc \leqq 0.2\ \mu g/g\ ^{235}U$

(2) Restriction of Contents

(i) Total Weight of Content	:	90 kg or less
(ii) Total Activity	:	6.60 GBq or less
(iii) Total Heat Generation Rate	:	Not applicable
(iv) Burnup Rate	:	Not applicable
(v) Cooling Time	:	Not applicable
(vi) Physical State	:	Solid (Powder and Pellet)
(vii) Maximum Contents per Package	:	See Attached Table

3. SPECIFICATION OF PACKAGING

(1) Total Weight of Packaging : 115 kg or less

(2) Outer Dimension of Packaging

(i) Outer Diameter	:	Approximately 61 cm
(ii) Height	:	Approximately 88 cm

(3) Materials of Packaging

(i) Inner Container

Inner Drum	:	Steel (SPCC or SPHC)
Flange and Lid	:	Steel (SS400)
Gasket	:	Butyl Rubber
Bolts and Nuts	:	Steel, (SS400)

(ii) Outer Container

Outer Drum	:	Steel (SPCC or SPHC)
Fastener	:	Steel (SS400)
Gasket	:	Natural Rubber
Heat Insulator	:	Pealite-Alumina Cement
Bolts and Nuts	:	Steel (SS400)

(4) Package Illustration : See Attached Figure

4. ASSUMED AMBIENT CONDITIONS

(i) Ambient Temperature : 38 ℃

(ii) Insolation Data : Table XII of IAEA Regulation (Safety Series No. 6, 1985 Edition)

5. RESTRICTIONS OF TRANSPORT

(i) Restriction Number : 500 Packages

(ii) Array : No Restriction

(iii) Transport Index for Nuclear Criticality Control : 0.1

6. SPECIAL FEATURES IN THE CRITICALITY ASSESSMENT

The subcriticality calculation is evaluated upon assumption that the cylinder is in immersion condition by water and no water leaks into the cylinder under the normal conditions and accident conditions in transport.

7. DECREASED NEUTRON MULTIPLICATION FACTOR

Any determination is not considered in the criticality assessment, because the subcriticality calculation is evaluated upon the condition of fresh nuclear fuels.

8. RESTRICTION ON THE MODES OF TRANSPORT

It is not confirmed that the design of package satisfies the additional requirements for packages transported by air.

9. INSTRUCTIONS ON USE AND MAINTENANCE OF PACKAGING

(1) Instructions on Maintenance of Packaging

The packaging shall be kept in good condition and required periodic inspections. Periodic inspections of each packaging shall be conducted more than once per year. (In case where a packaging is used for transport more than ten times per year, the periodic inspections shall be conducted at least once every ten transports.)

The periodic inspections shall include visual inspection for gaskets of each packaging and subcriticality inspection.

The packages or packaging shall be lifted with a forklift or crane.

(2) Actions prior to Shipment

Each package shall be checked for the following items before shipments.

(i) Visual Inspection
(ii) Lifting Inspection
(iii) Weight Measurement

(iv) Surface Contamination Measurement
(v) Radiation Dose Rate Measurement
(vi) Subcriticality Inspection
(vii) Content Inspection

Gaskets of the packaging shall be inspected, and exchanged if necessary.

(3) Precautions for Loading of Package for Transport

Loading of the package shall be performed such that the package will not move, roll down or fall down during transport.

10. THE ISSUE DATE AND EXPIRY DATE OF CERTIFICATE

(1) Issue Date	:	August 8, 1995
(2) Expiry Date	:	August 8, 1998

11. NOTE

This certificate does not relieve the consignor from compliance with any requirement of the government of any country through or into which the package will be transported.

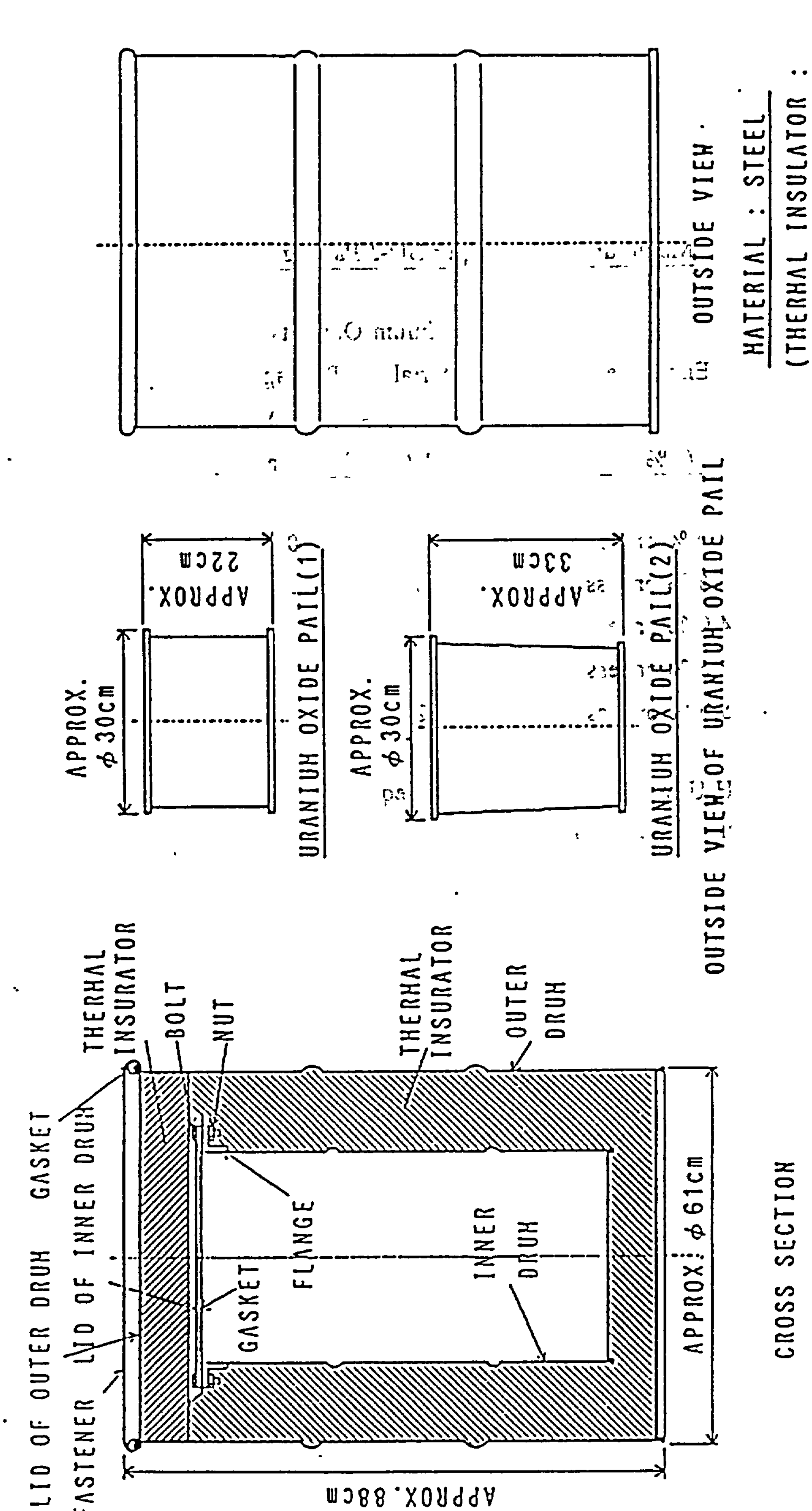
Attached Table

Maximum Contents per BU-J Package

Enrichment (%)	Maximum Qunatity of Material per Package (kg - UO_2) Pellet	Powder
3.0 % or less	76.2	89.0
3.6 % or less	57.0	62.2
4.0 % or less	49.4	51.4
4.6 % or less	40.0	40.4
5.0 % or less	36.2	36.2

H/U ratio shall be confirmed equal or less than 0.45 prior to shipment.

Attached Figure



OUTSIDE VIEW OF BU-J PACKAGE